3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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," Revision 1.

> In this chapter, the NRC staff uses the term "nonsafety-related" to refer to structures, systems and components (SSCs) that are not classified as "safety-related SSCs" as described in 10 CFR 50.2. However, among the nonsafety-related" SSCs, there are those that are "important to safety" as that term is used in the General Design Criteria (GDC) listed in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, and others that are not considered "important to safety."

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) of 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."

Chapter 3, "Design of Structures, Components, Equipment, and Systems," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), begins with Section 3.2.1, "Seismic Classification"; therefore, the staff did not perform a specific review of DCA Part 2, Tier 2, Section 3.1. In general, each individual section of this chapter discusses conformance to the applicable GDC.

3.2 Classification of Structures, Systems, and Structures

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 defined 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. As defined in 10 CFR Part 50, Appendix S, the safety-related SSCs required to withstand the effects of the safe-shutdown earthquake (SSE) ground motion are those necessary to assure one of the following:

- the integrity of the reactor coolant pressure boundary (RCPB)
- the capability to shut down the reactor and maintain it in a safe-shutdown condition

- 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 SRP Section 3.2.1, which references RG 1.29. The objective of the staff's review is 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 on 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. 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 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.

3.2.1.3 Regulatory Basis

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

- GDC 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 and the applicable quality assurance (QA) requirements of 10 CFR Part 50, Appendix B, as they relate to applying QA requirements to activities that affect the safety-related functions of SSCs designated as seismic Category I, commensurate with their importance of the safety functions to be performed.

- GDC 2, “Design Bases for Protection against Natural Phenomena,” of Appendix A to 10 CFR Part 50, 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 61, “Fuel Storage and Handling and Radioactivity Control,” of Appendix A to 10 CFR Part 50, as it relates to the design of radioactive waste systems and other systems that may contain radioactivity to assure adequate safety under normal and postulated accident conditions.

- Appendix A, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” to 10 CFR Part 100, “Reactor Site Criteria,” and Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” to 10 CFR Part 50, as it relates 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 the acceptance criteria adequate to meet the above requirements and provides 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 1.29 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

In GDC 2, the NRC requires, in part, 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)

RG 1.29, Revision 4, Section C, states that the following SSCs of a nuclear power plant, including its foundations and supports, should be designated as seismic Category I systems or portions thereof (including, but not limited to, systems such as residual heat removal and auxiliary feedwater (FW)) that are needed to do one of the following:

- Shut down the reactor and maintain it in a safe-shutdown condition.
- Remove residual heat (including heat stored within the spent fuel pool).
- Control the release of radioactive material.
- Mitigate the consequences of an accident.

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, except for a few components, as described below.

The column listed “N/A” for ultimate heat sink (UHS) pool (SSC Class A1) in DCA Part 2, Tier 2, Table 3.2-1, as supplemented by the applicant’s letter dated October 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17290B266), does not identify a seismic classification for UHS pool. The staff requested the applicant provide the seismic classification for the UHS pool in DCA Part 2, Tier 2, Table 3.2-1. In its supplemental letter, the applicant clarified that the water of the UHS pool is classified to be “N/A” as shown in a markup to Table 3.2-1. The UHS-related components are classified as seismic Category I. DCA Part 2, Tier 2, Revision 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. After reviewing this information, 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.

DCA Part 2, Tier 2, Table 3.2-1, does not completely list the classification of the reactor vessel internals (RVIs). As supplemented by letter dated October 17, 2017 (ADAMS Accession No. ML17290B266), the applicant indicated that the RVIs are designed to meet seismic Category I requirements to withstand the effects of earthquakes without loss of capability to perform their safety-related functions. 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. 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. RG 1.29, Staff Regulatory Position C.1.b, states that the reactor core and RVIs are the nuclear power plant SSCs, including its foundations and supports, that should be designated as seismic Category I. Based on this review, the staff finds that the classification of RVIs conforms to RG 1.29, Staff Regulatory Position C.1.b, and concludes that the classification of RVIs is acceptable.

DCA Part 2, Tier 2, Table 3.2-1, Note 2, states that “AQ-S indicates that the pertinent requirements of 10 CFR 50 Appendix B are applicable to SSC classified as Seismic category II in accordance with the quality assurance program.” However, several components labeled as “AQ-S” are designed to seismic categories that are not seismic Category II. RG 1.29, Staff Regulatory Position C.1, states 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.

As supplemented by letter dated October 17, 2017 (ADAMS Accession No. ML17290B266), the applicant stated that it revised the bullet in Note 2 for "AQ-S" to clarify that nonsafety-related, seismic Category I SSCs also meet the pertinent requirements of 10 CFR Part 50, Appendix B, in accordance with the QAP. With this clarification, the pertinent requirements of 10 CFR Part 50, Appendix B, are correctly identified as being applicable to both seismic Category I and II, nonsafety-related SSCs designated as "AQ-S" in DCA Part 2, Tier 2, Table 3.2-1. "AQ-S" is intended to be applied to different seismic categories to account for nonsafety-related SSCs that can either be seismic Category I or seismic Category II. The staff finds that the revised Note 2 of DCA Part 2, Tier 2, Table 3.2-1, conforms to RG 1.29, Staff Regulatory Position C.1, and concludes that the classification of "AQ-S" is acceptable.

As part of the updates to DCA Part 2, Tier 2, Table 3.2-1, the applicant added, "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 Section 3.7.3.8 ["Interaction of Non-Seismic Category I Subsystems with Seismic Category I SSC"]." The staff finds that the addition of Note 5 to Table 3.2-1 is acceptable because it conforms to RG 1.29. SER Section 3.7.3.8 describes the additional safety evaluations of the effects of seismic Category III structures on seismic Category I structures in detail.

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 DCA Part 2, Tier 2. 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 the design information described in DCA Part 2, Tier 2, was adequately translated into the classification documentation. 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." As a follow-up to the audit observations, the NRC requested the applicant provide 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.

In DCA Part 2, Tier 1 and Tier 2, as supplemented by letter dated December 17, 2018 (ADAMS Accession No. ML18351A357), the applicant provided the changes to 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 the revised DCA Part 2, Tier 2, Figure 6.6-1, “ASME Class Boundaries for NuScale Power Module Piping Systems.” The applicant also referenced the revised figure in its discussion of ITAAC 02.01.01 and ITAAC 02.02.01 in DCA Part 2, Tier 2, Section 14.3. The changes included component/system class break information. The staff finds the changes acceptable because the class break information conforms to RG 1.29. The NRC staff is tracking the applicant’s letter dated December 17, 2018 (ADAMS Accession No. ML18351A357), as **Confirmatory Item 03.02.01-1**, until the applicant includes the markups of DCA Part 2, Tier 1, Tables 2.1-1 and 2.1-2 and revised Figure 6.6-1, in the next revision of the DCA.

DCA Part 2, Tier 2, Revision 0, Table 3.2-1, classifies the bioshields as nonsafety-related, not risk-significant, seismic Category II components; DCA Part 2, Tier 2, Table 3.2-1, identifies them as B2. DCA Part 2, Tier 2, Table 3.2-1, does not provide the safety classification information for the bioshield that would allow the staff to conclude that it is appropriately classified. Subsequently, as supplemented by letter dated November 16, 2018 (ADAMS Accession No. ML18320A254), the applicant redesigned the bioshield to have vent paths that allow venting, which eliminates the need for movement of a relief panel. The bioshield vents are no longer mechanical devices; the hinges have been removed. 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 (nonsafety-related and nonrisk significant) seismic Category II. The staff finds the classification acceptable because the classification of bioshield conforms to RG 1.29. In its letter dated November 16, 2018 (ADAMS Accession No. ML18320A254), the applicant provided the design changes of the bioshield vents that will be incorporated in the next revision of DCA Part 2, Tier 1, Section 3.11, “Reactor Building,” and Table 3.11-1, “Reactor Building Shield Wall Geometry,” and DCA Part 2, Tier 2, Section 1.2.2.1, “Reactor Building”; Section 3.7.2, “Seismic System Analysis”; Section 3.7.3.3, “Procedures Used for Analytical Modeling”; and Section 12.3, “Radiation Protection Design Features.” Until the changes are incorporated into the next revision of DCA, the NRC staff is tracking this as **Confirmatory Item 03.02.01-2**.

SER Sections 1.2, 3.7.2, 3.7.3, and 12.3 describe the additional safety evaluations of the new bioshield vent design in detail.

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

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 module 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, pending the completion of **Confirmatory Items 03.02.01-1** and **03.02.01-2**. In addition, the staff finds that DCA Part 2, Tier 2, includes an acceptable process to meet RG 1.29, Staff Regulatory Positions C.2, C.3, and C.4. This constitutes an acceptable basis for satisfying, the portions of 10 CFR Part 50, Appendix A, GDC 1, GDC 2, and GDC 61; 10 CFR Part 100, Appendix A; 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, Revision 4, issued March 2007.

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, Revision 4, 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. SER Section 5.2.1.1 discusses the conformance of RCPB components to the requirements of 10 CFR 50.55a, "Codes and Standards." RG 1.26 designates these RCPB 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 NUREG-0800 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 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 Code, Section III. In accordance with 10 CFR 50.55a(e)(1), QC C components must meet the requirements for Class 3 components in ASME Code, Section III.

3.2.2.2 Summary of Application

DCA Part 2, Tier 1: 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: 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. Nonsafety-related fluid systems and components 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 Class A1 and determined to be both safety related and risk significant, A2 designates SSCs Class A2 and determined to be both safety related and not risk significant, Class B1 designates SSCs that are determined to be both nonsafety-related and risk-significant, and Class B2 designates SSCs that are determined to be both nonsafety-related and not risk significant. Certain nonsafety-related SSCs that perform risk-significant functions 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 nonsafety-related SSCs for which specific regulatory guidance applies (e.g., RG 1.29). DCA Part 2, Tier 2, Table 3.2-1, identifies the application of 10 CFR Part 50, Appendix B, to specific nonsafety-related SSCs.

DCA Part 2, Tier 2, Section 3.2, "Classification of Structures, Systems, and Components," states that the classification methodology includes consideration for "augmented" requirements for those SSCs that are, by definition, nonsafety-related (based on 10 CFR 50.2, "Definitions"). The applicant states that the selection of augmented requirements is based on a consideration of the important functionality to be performed by the nonsafety-related SSCs and on regulatory guidance applicable to the functionality (e.g., consistent with the functionality specified in GDC 60, "Control of Releases of Radioactive Materials to the Environment," 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 structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

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.

3.2.2.3 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 they relate to components that are part of the RCPB that must meet the requirements for Class 1 components in ASME Code, Section III, except as provided in 10 CFR 50.55a(c)(2) through (4).
- 0 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 Code, Section III.
- 10 CFR 50.55a(e)(1) as it relates to Quality Group C components that must meet the requirements for Class 3 components in ASME Code, Section III.

SRP Section 3.2.2 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, 10 CFR 50.55a(e) for QG C. The guidance in RG 1.26 is used to establish the quality groups for other safety-related components that contain water, steam, or radioactive material. DCA Part 2, Tier 2, Table 1.9-2, “Conformance with Regulatory Guides,” indicates that the applicant’s 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 technical evaluations.

The staff reviewed DCA Part 2, Tier 2, Section 3.2.2, and provided the technical evaluations discussed below.

DCA Part 2, Tier 2, Table 3.2-1, indicates that no equipment is considered for the RTNSS. As supplemented by letter dated October 17, 2017 (ADAMS Accession No. ML17290B241), the applicant assessed the risk significance of the SSCs with respect to initiating frequencies. There is no equipment considered for the classification of RTNSS components. Because the applicant is not relying on the function of nonsafety-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. Therefore, this issue is resolved and closed.

The column "Augmented Design Requirements," in DCA Part 2, Tier 2, Table 3.2-1, does not completely specify codes and standards for all components (e.g., DCA Part 2, Tier 2, Table 3.2-1, does not clearly present the codes and standards to be used for the design, fabrication, erection, and testing of components for the auxiliary boiler system (ABS); QG D; and the RVIs). As supplemented by letter dated October 17, 2017 (ADAMS Accession No. ML17290B241), the applicant provided the following changes to DCA Part 2, Tier 2, Table 3.2-1:

- The applicant revised Note 3 for the column "Augmented Design Requirements" of DCA Part 2, Tier 2, Table 3.2-1, to clarify that additional augmented design requirements are reflected in the columns "Quality Group/Safety Classification" and "Seismic Classification."
- The applicant revised Note 4 and expanded DCA Part 2, Tier 2, Section 3.2.2.4, "Quality Group D," to identify the codes and standards for SSCs designated as QG D in accordance with RG 1.26.
- The applicant relocated the reference of RG 1.143 in Note 2 and deleted QG D of the hydraulic skid for the valve reset.
- The applicant revised the "AQ" bullet in Note 2 of Table 3.2-1 to clarify the applicability of augmented QA requirements for nonsafety-related SSCs.
- 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."
- The applicant revised DCA Part 2, Tier 2, Table 3.2-1, and, for the RVIs, listed SG supports separately as components of the reactor coolant system (RCS).
- The applicant revised DCA Part 2, Tier 2, Table 10.4-20, "Auxiliary Boiler System Component Design Parameters," as supplemented by letter dated September 18, 2017 (ADAMS Accession No. ML17261B280), to identify the section of the ASME Code that applies to the fired power boiler.

The staff finds that the changes discussed in the above paragraphs are acceptable because they conform to RG 1.26 and RG 1.143. Therefore, the issue is resolved and closed.

The column "Augmented Design Requirements," in DCA Part 2, Tier 2, Revision 0, Table 3.2-1, does not completely define the treatment and design requirements of component supports (i.e., SG tube supports with QG A, “none” in the “Augmented Design Requirements” column, reactor pressure vessel (RPV) support stand with QG C, “none” in the “Augmented Design Requirements” column).

As supplemented in its letter dated October 17, 2017 (ADAMS Accession No. ML17290B241), the applicant made the following changes to DCA Part 2, Tier 2, Revision 2, Table 3.2-1, and provided the following explanations:

- The applicant revised DCA Part 2, Tier 2, Section 3.2.2, to provide the requirements for the supports for the SSCs that meet each QG classification in RG 1.26. To clearly provide the design requirements for supports for SSCs in QG A, B, C, and D, the applicant provided changes to DCA Part 2, Tier 2, Sections 3.2.2.1 through 3.2.2.4, to 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.

- The applicant revised DCA Part 2, Tier 2, Sections 3.2.2.1 through 3.2.2.4, to include the classification information of 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 within the ASME BPV Code and, therefore, complies with 10 CFR 50.55a(c), (d) and (e).

The staff finds that the changes discussed in the above paragraphs conform to RG 1.26 and are acceptable. Therefore, the issue is resolved and closed.

DCA Part 2, Tier 2, Revision 0, 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. However, neither section discusses the threaded fasteners, which are part of the RCPB.

As supplemented by letter dated October 17, 2017 (ADAMS Accession No. ML17290B241), the applicant included the fastener classification in DCA Part 2, Tier 2, Section 5.2.1, Section 5.2.4.1, and Table 3.2-1. 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 Code, Section III and SRP Section 3.2.2 and is, therefore, acceptable.

DCA Part 2, Tier 2, Section 10.3.6.2, “Materials Selection and Fabrication,” states that Section 3.2 provides the material specification, grade, and classification for piping, valves, fittings, and weld filler material used in the main steam system (MSS) and condensate and feedwater system (CFWS). DCA Part 2, Tier 2, Section 10.3.6.2, also states that the MSS and CFWS conform to ASME BPV B31.1, “Power Piping,” and are consistent with the QG and seismic design classifications in DCA Part 2, Tier 2, Revision 0, Table 3.2-1. However, DCA Part 2, Tier 2, Revision 0, Table 3.2-1, did not present any of this information.

As supplemented by its letter dated June 26, 2017 (ADAMS Accession No. ML17177A686), the applicant stated that the changes were the result of a May 16, 2017, public teleconference with the staff. DCA Part 2, Tier 2, Section 10.3.6.2, no longer refers to DCA Part 2, Tier 2, Section 3.2, for material specification and grade. DCA Part 2, Tier 2, Table 10.3-5, “Material Specifications and Corrosion Allowances,” was added to provide material specification and grade along with corrosion allowances. DCA Part 2, Tier 2, Section 10.3.6.2, now refers to DCA Part 2, Tier 2, Table 3.2-1, only for the QG and seismic design classifications. The applicant added the codes and standards for QG D SSCs to DCA Part 2, Tier 2, Section 3.2.2.4. The staff finds that the changes described above conform to RG 1.26 and are acceptable.

DCA Part 2, Tier 2, Revision 0, Table 3.2-1, did not completely provide the classification information on the piping systems of the steam generator system (SGS), decay heat removal system (DHRS), ABS, CFWS, and UHS. As supplemented by letter dated October 17, 2017 (ADAMS Accession No. ML17290B241), the applicant clarified that 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 applicant corrected the QG classification for these components to be QG B, as identified in DCA Part 2, Tier 2, Section 5.4.3.1, “Design Basis.” Accordingly, the applicant has revised DCA Part 2, Tier 2, Table 3.2-1, to correct the QG designation. The staff finds that the classification approach of piping components and the changes to 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 with 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, and other miscellaneous handling equipment). In addition, the P&IDs in the DCA identify the classification boundaries of interconnecting piping and valves. The DCA states (or demonstrates?) that all the above SSCs will be constructed in conformance with applicable ASME BPV Code and industry standards. 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, 10 CFR Part 50, Appendix A, GDC 1, therefore, is 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 wind speed, 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,” 2005, which is a reference of practice in wind design.

For the NuScale design, the applicant used a basic wind speed of 233 kilometer per hour (145 miles per hour (mph)), a 100 year return period, a 3 second gust at 10 meters (33 feet) aboveground 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: DCA Part 2, Tier 1, Chapter 3, provides the ITAAC associated with DCA Part 2, Tier 2, Section 3.3.1.

Technical Specifications: There are no TS 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, hurricane, 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.

3.3.1.4 Technical Evaluation

SER Sections 2.3.1 and 2.3.2 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, which 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 wind speed 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 wind speed 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, "Wind Loadings." 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.

Table 3.3.1-1 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, "Extreme Wind Loads (Tornado and Hurricane Loads)."

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 RXB or Seismic Category I portion of the CRB.	3.3.1

3.3.1.6 Conclusion

The staff concludes that the severe wind loadings 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, and the staff has determined that the information in DCA Part 2 provides a reasonable assurance of safety 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, 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, "Site Design Parameters," 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 wind speed 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 wind speed 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 of the individual tornado and hurricane loading components and load factors.

For the NuScale design, the applicant used the maximum design-basis tornado wind speed of 230 mph with the translational speed of 46 mph, the maximum rotational speed of 184 mph, the radius of maximum rotational speed of 150 feet, the pressure drop of 1.2 per square inch (psi), and the rate of pressure drop 0.5 psi per second for the design of the RXB, CRB, and RWB. In addition, the applicant used the maximum design-basis hurricane wind speed of 290 mph for the design of these buildings.

ITAAC: DCA Part 2, Tier 1, Chapter 3, provides the ITAAC associated with DCA Part 2, Tier 2, Section 3.3.2.

Technical Specifications: There are no TS 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, hurricane, 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.

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 wind speed 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 Sections 3.5.1 and 3.5.2 document 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.	3.3.2

3.3.2.6 Conclusion

The staff concludes that the extreme wind loadings used in the design of the SSCs for the NuScale application are acceptable and meet the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2, and the staff has determined that the information presented in DCA Part 2 provides a reasonable assurance of safety to withstand the effects of the tornado and hurricane phenomena.

3.4 <u>**Water Level (Flood) Design**</u>

3.4.1 Internal Flood Protection for Onsite Equipment Failure

3.4.1.1 Introduction

The NRC staff reviewed DCA Part 2, Tier 1, Section 3.11.1 and Section 3.13.1, 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." DCA Part 2, Tier 2, Section 3.4.1, states that the internal flooding analyses were performed for the NuScale RXB and CRB to confirm that internal flooding from postulated failures of tanks, piping, or actuation of fire suppression system does not cause a loss of SSCs that are 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.

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. 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 requiredto (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.

Technical Specifications: There are no proposed TS requirements associated with internal flood protection.

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

3.4.1.3 Regulatory Basis

SRP Section 3.4.1, Revision 3, issued March 2007, provides the relevant regulatory requirements for this area of review and the associated acceptance criteria, summarized below, as well as review interfaces with other SRP sections:

- 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, tornados, 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).

- 10 CFR 52.47(b)(1), as it requires a DCA to contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the AEA, and the NRC's rules and regulations.

3.4.1.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 1, Section 3.11.1, "Design Description," and Section 3.13.1, "Design Description," and DCA Part 2, Tier 2, Section 3.4.1, in accordance with SRP Section 3.4.1 to ensure compliance with the guidance 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 the NuScale application, the applicant conducted a level-by-level and room-by-room flooding analysis consisting of the following steps:

- Identify potential flooding sources.

- Identify rooms/areas that contain equipment subject to flood protection.

- Estimate the flood depth in the identified rooms/areas.

- Determine the need for protection and mitigation measures for rooms containing equipment subject to flood protection.

DCA Part 2, Tier 2, Tables 3.4-1 and 3.4-2, give 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/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. A COL applicant that references the NuScale DC will confirm the final location of SSCs subject to flood protection, select the mitigation strategy, develop an inspection and maintenance program, and confirm that site-specific tanks or water sources are located where they cannot cause adverse flooding conditions in the RXB and CRB.

However, the staff's review of the application identified inconsistencies between the assumptions used in the internal flooding analyses and RG 1.189 with respect to the available amount of water in the fire suppression system and the duration of the fire suppression activity. For this reason, the staff issued **RAI 9052**, **Question 03.04.01-1** (ADAMS Accession No. ML17224A023), and **RAI 9053**, **Question 03.04.01-2** (ADAMS Accession No. ML17224A024), and followed up with **RAI 9330**, **Question 03.04.01-3** (ADAMS Accession No. ML18003B378), and **RAI 9331**, **Question 03.04.01-4** (ADAMS Accession No. ML18003B379), to ask the applicant to confirm that the assumptions in the flooding analyses are bounding conditions for all areas in the RXB and CRB or to provide justifications for the reduced flow rate and fire suppression activity time duration.

In its response dated March 5, 2018 (ADAMS Accession No. ML18064A889), to **RAI 9330**, **Question 03.04.01-3**, the applicant stated that fire barriers divide the RXB and CRB into fire areas and concluded that 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 0.3 gallon per minute (gpm) for the RXB and 0.2 gpm for the CRB. In addition, the applicant stated that the fire suppression flow estimate for each fire area includes one manual hose stream of 250 gpm. Considering the methodology used by the applicant to perform the flooding analysis, which established that fire areas are compartmentalized and limited to 1,500 ft^2, the staff concluded that the assumptions of fire suppression activity discharges of 700 gpm for the RXB and 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 adequate. These assumptions 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."

In the internal flooding analysis, the applicant also assumed that water flows through pipe ruptures are limited to 40 and 30 minutes in the RXB and CRB, respectively. The staff requested the applicant to justify the above assumptions in **RAI 9053**, **Question 03.04.01-2** (ADAMS Accession No. ML17224A024). In its response dated October 11, 2017 (ADAMS Accession No. ML17284A914), to **RAI 9053**, **Question 03.04.01-2**, the applicant stated that "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 walk-downs, use of plant monitoring equipment, and video monitoring systems as described above, which is consistent with the guidance in SRP Section 3.4.1. RAI 9053 is resolved.

Because the timely isolation of ruptured piping depends on more than just the prompt detection of leakages, the staff issued followup **RAI 9331**, **Question 03.04.01-4** (ADAMS Accession No. ML18003B379), requesting the applicant to describe any other design features such as the availability and accessibility of isolation valves, limited water volumes, or credited actions to isolate postulated ruptured piping, and if the COL applicants are to develop specific mitigation strategies, the applicant should provide a COL information item, as appropriate. In its response dated August 24, 2018 (ADAMS Accession No. ML18236A893), to **RAI 9331**, **Question 03.04.01-4**, the applicant proposed adding COL Item 3.4-2, which states, "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." In addition, the applicant will revise DCA Part 2, Tier 2, Section 3.4.1.5 and Table 1.8-2 as described above. The staff finds the applicant's response to **RAI 9331**, **Question 03.04.01-4**, acceptable because 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. The staff confirmed that the applicant included COL Item 3.4-2 in DCA Part 2; therefore, **RAI 9331**, **Question 03.04.01-4**, is resolved and closed.

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 environment effects of pipe breaks. The information provided in the DCA is consistent with the guidance in SRP Section 3.4.1 by comprehensively identifying flooding hazards caused by potential line break accidents and fire suppression activities and provided appropriate mitigation measures, where needed, to preclude adverse effects on safety-related equipment and SSCs important to safety. The staff also determined that the applicant has provided appropriate ITAAC in DCA Part 2, Tier 1, Tables 3.11-2 and 3.13-1, to afford reasonable assurance that the design commitments will be met and that the as-built plant will operate in accordance with the DC.

3.4.1.5 Inspections, Tests, Analyses, and Acceptance Criteria

DCA Part 2, Tier 1, Tables 3.11-2 and 3.13-1, prescribe ITAAC related to internal flood protection for the RXB and CRB, respectively. These ITAAC ensure that barriers, including flood-resistant doors, curbs and sills, walls, watertight penetration seals, and National Electrical Manufacturers Association enclosures, exist and are qualified in accordance with the internal flooding analysis to provide confinement so that the impact from an internal flood in the RXB or CRB is contained within the flooding area of origin.

3.4.1.6 Initial Test Program

No initial tests requirements are associated with internal flood protection. The system is nonsafety-related and is not required for safe shutdown; therefore, no initial test requirements are needed.

3.4.1.7 Technical Specifications

No TS requirements are associated with internal flood protection. The system is nonsafety-related and is not required for safe shutdown; therefore, no TS requirements are needed

3.4.1.8 Combined License Information Items

SER Table 3.4.1 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

COL 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

3.4.1.9 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, with respect to accommodating the effects of internal floods.

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, “Site Design Parameters,” provides the Tier 1 information associated with this section.

DCA Part 2, Tier 2: The applicant provided the flood and ground water site design 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 1 foot below the baseline plant elevation (100’–0”) and the maximum ground water elevation for the design is 2 feet below the baseline plant elevation. In addition, the applicant described the bounding parameters for both rain and snow and the design features necessary to protect the safety-related and risk-significant SSCs from ground water intrusion without the use of a permanent dewatering system. The applicant described the analysis procedures that are used to transform the static effects of the highest flood and ground water levels into effective loads applied to seismic Category I structures.

ITAAC: DCA Part 2, Tier 1, Chapter 3, provides the ITAAC associated with DCA Part 2, Tier 2, Section 3.4.2.

Technical Specifications: There are no TS 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, hurricane, 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.

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 design 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 only needs to consider 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, "Analysis Procedures," 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 addressed 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, applicant proposed inclusion of COL Item 3.4-5 and COL Item 3.4-7, which will require 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 requires 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 the COL items 3.4-5 and 3.4-7 and the COL items are 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 item to be appropriate for this case, as it serves to remind the COL applicant 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 staff documents its evaluations of interaction of non-Seismic Category I structures with seismic Category I Structures. The applicant stated that the seismic Category II 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 Table 1.8-2
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 Table 1.8-2
COL Item 3.4-7	A COL applicant that references the NuScale Power Plant design certificate will determine the extent of waterproofing and dampproofing needed to prevent groundwater and foreign material intrusion into the expansion gap between the end of the tunnel between the RXB and the CRB, and the corresponding RXB connecting walls.	3.4.2.1 Table 1.8-2

3.4.2.6 Conclusion

The staff concludes that the NuScale design, as it relates to protection of structures against flood from external sources is acceptable and meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2, and the staff has determined that the information presented in DCA Part 2 provides reasonable assurance of safety to withstand the effects of the highest flood and ground water levels specified for the NuScale DC.

3.5 <u>Missile Protection</u>

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).”

SRP Section 3.5.1.1 delineates 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. 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 1, Sections 2.1, 3.11.1, and 3.13.1, describe the design of the NuScale Power Module (NPM), RXB, and CRB, respectively.

DCA Part 2, Tier 2: 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: DCA Part 2, Tier 1, Table 2.1-4, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria"; Table 3.11-2; and Table 3.13-1, provide ITAAC requirements for the NPM, RXB, and CRB, respectively.

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

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

3.5.1.1.3 Regulatory Basis

SRP Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)," and SRP Section 3.5.1.2, "Internally-Generated Missiles (Inside Containment)," provide the relevant regulatory requirements for this area of review and the associated acceptance criteria, summarized below, as well as the review interfaces with other SRP sections:

- 10 CFR Part 50, Appendix A, 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 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.

- 10 CFR 52.47(b)(1), as it requires a DCA to contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC has been constructed and will operate in accordance with the DC, the provisions of the AEA, and the NRC's rules and regulations

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. 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 the following RGs:

- RG 1.115, “Protection against Low-Trajectory Turbine Missiles,” Revision 1, issued July 1977, Staff Regulatory Positions C.1 and C.3, as they relate to the protection of the SSCs important to safety from the effects of turbine missiles

  Regulatory Position C.1 specifies that essential systems of a nuclear power plant should be protected against low-trajectory turbine missiles caused by failure of main turbine generator sets. Consideration may be limited to the SSCs listed in Appendix A to RG 1.117, “Tornado Design Classification,” Revision 1, issued April 1978. The effect of physical separation of redundant or alternative systems may also be considered. Each essential system and its location should be identified on dimensioned plan and elevation layout drawings. 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.

- RG 1.117, Revision 1, Appendix A, as to which SSCs should be protected from missile impacts

DCA Part 2, Tier 2, Section 3.5, “Missile Protection,” 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 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, if the product of missile generation probability (P_1), missile impact probability (P_2), and damage probability (P_3) is less than $1x10^{-7}$ per year, the missile is not considered statistically significant. If the product is greater than $1x10^{-7}$ per year, barriers or other measures are taken to protect the SSCs.

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:

- moderate- and low-energy systems with operating pressures of less than 275 pounds 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 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 ASME BPV Code limiting stresses in the bonnet-to-body bolting material or using a retaining ring prevents bolted bonnet valves and pressure-seal bonnet valves constructed to ASME BPV Codes and standards from becoming missiles. However, the applicant did not identify the specific ASME BPV Code or standard that would demonstrate a high level of quality and thus assure the structural integrity of the valves in order to conclude that the missile sources are not considered credible. Therefore, the staff issued **RAI 8770**, **Question 03.05.01.01-1** (ADAMS Accession No. ML17130A825), requesting the applicant to provide additional technical justification or specific ASME BPV Codes and standards that demonstrate that these valves should not be considered credible missiles.

The its response to **RAI 8770**, **Question 03.05.01.01-1**, the applicant stated that it will revise DCA Part 2, Tier 2, Section 3.5.1.1.1, to provide the codes and standards that are applied to bolted bonnet valves and pressure-seal bonnet valves. Specifically, 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.

The staff reviewed the applicant’s response to **RAI 8770**, **Question 03.05.01.01-1**, and finds it acceptable because the specification of design and construction codes demonstrates the structural integrity of the components and minimizes the likelihood of missile generation. DCA Part 2 incorporates the markups provided in **RAI 8770**, **Question 03.05.01.01-1**; therefore, **RAI 8770**, **Question 03.05.01.01-1**, is resolved and closed. 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 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 3.5.1.1.2, 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 are seismically restrained, are removed from the building, or are moved to a location where they are 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.

As part of its review of hazards associated with internally generated missiles, 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 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 conclusions on the elimination of the above components as credible missile sources, 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 SER Section 15.4.8 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 Inspections, Test, Analyses, and Acceptance Criteria

The NPM, RXB and CRB buildings are included in the ITAAC as described in DCA Part 2, Tier 1, Table 2.1-4, “NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria”; Table 3.11-2; and Table 3.13-1, which provide ITAAC requirements for the NPM, RXB, and CRB being seismic Category I and missile protection. The staff finds that the seismic Category 1 requirements in the ITAAC associated with the NPM, RXB, and CRB are sufficient to provide reasonable assurance that safety-related SSCs and risk-significant SSCs requiring missile protection will be protected from internally generated missiles. The NRC staff’s evaluation of these three ITAAC items are addressed in SER Section 14.3.

3.5.1.1.6 Combined License Information Items

No COL information items are directly associated with this review area. The staff finds that DCA Part 2, Tier 2, Table 1.8-2, does not need to include any additional COL information items for protection from internally generated missiles.

3.5.1.1.7 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 with the guidance in SRP Section 3.5.1.1 with regard to which SSCs should be protected from missile impacts.

3.5.1.2 Internally Generated Missiles (Inside Containment)

SER Section 3.5.1.1 evaluates internally generated missiles inside containment.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Introduction

In GDC 4, the NRC requires SSCs important to safety to be appropriately protected against dynamic effects of postulated accidents, 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. One potential source of plant missiles is the rotor of the main turbine. As such, 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 are 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 letters dated December 21, 2017 (ADAMS Accession No. ML17355A168), June 25, 2018 (ADAMS Accession No. ML18176A394), October 29, 2018 (ADAMS Accession No. ML18302A411) and October 31, 2018 (ADAMS Accession No. ML18304A306) describe the NuScale DC, as summarized, in part, below.

DCA Part 2, Tier 2, Figure 1.2-2, shows the turbine generator building layout. 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, issued January 2012.

In accordance with its letters dated December 21, 2017, June 25, 2018, October 29, 2018 and October 31, 2018, the applicant revised DCA Part 2 to use barriers as the basis for justifying adequate protection from turbine.

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.

3.5.1.3.3 Regulatory Basis

“Design-Specific Review Standard for the NuScale SMR Design” (DSRS), Section 3.5.1.3, “Turbine Missiles,” Revision 0, issued June 2016, 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 10 CFR Part 50, Appendix A, 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 represent 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 guidance in RG 1.115, Revision 2, issued January 2012, applies as it relates to the identification of low-trajectory missiles resulting from turbine failure.

- 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,” (ADAMS Accession No. ML101970069) issued July 1965, 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. The probability of a strike by a turbine missile should be sufficiently low so that the risk from turbine missiles to essential SSCs is acceptably small or essential SSCs should be protected using barriers. DCA Part 2, Tier 2, Section 3.5.1.3, provides information on determining the orientation of the turbine generator with respect to essential SSCs. 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 orientated such that essential SSCs, including the NPMs, are within the low-trajectory turbine missile strike zone, as defined by RG 1.115. The staff concludes that the turbine generators are unfavorably orientated as defined by RG 1.115, based on the plant layout with the RXB and CRB. Therefore, barriers are necessary to protect essential SSCs in the RXB and CRB.

By its letter dated June 25, 2018, (ML18176A394), the applicant revised DCA Part 2 sections concerning turbine missiles by deleting Section 10.2.3, “Turbine Rotor Integrity” of the DCA Part 2 and modifying Section 3.5.1.3 of the DCA Part 2 to use barriers in lieu of turbine rotor integrity and probabilistic analysis as the bases for providing adequate protection from turbine missiles.

On July 16, 2018, the staff conducted a public meeting to discuss the applicant’s revised methodology of using barriers to protect essential SSCs from turbine missiles. The staff issued RAI 9596, Question 03.05.03-4, Subpart (1) (ML18250A323) based on this meeting to request that the basis and supporting documentation be provided for concluding that barriers are adequate to protect the essential SSCs from a spectrum of turbine missiles that includes up to half of the last stage of the rotor with the blades attached. In its letter dated October 31, 2018 (ML18304A306), the applicant did not provide the basis and supporting documentation, including the supporting analyses for the full spectrum of turbine missiles up to half of the last stage of the rotor with the blades attached. Therefore, the staff conducted an audit to review pertinent technical information including the analyses, calculations, engineering drawings, design assumptions and the technical bases for the turbine missile parameters (mass, size and velocity), and verification that the structural missile 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. Based on the audit, the staff determined that the applicant has not performed an analysis for the full spectrum of turbine missiles which includes up to half of the last stage of the rotor with the blades attached. Therefore, the staff is tracking **RAI 9596**, **Question 03.05.03-4, Subpart (1)**, concerning the spectrum of turbine missile as **Open Item 03.05.01.03-1**. See Section 3.5.3 of this SER concerning the barriers used to protect essential SSCs, which is being tracked as **Open Item 03.05.03-1.**

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 Item 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 certified design will provide a missile analysis for the turbine generator which demonstrates that the probability of a turbine generator producing a low trajectory missile is less than 10^{-5}.	3.5.1.3
COL Item 3.5-2	A COL applicant that references the NuScale Power Plant certified design will address the effect of turbine missiles from nearby or co-located facilities.	

3.5.1.3.6 Conclusion

Given **Open Item 03.05.01.03-1** described above, the staff cannot determine whether the NuScale information for turbine missiles is acceptable because the applicant has not provided sufficient information as it relates to the use of barriers.

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, “Site Design Parameters,” 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 TS 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

SRP Section 3.5.1.4, “Missiles Generated by Tornadoes and Extreme Winds,” provides the relevant regulatory requirements for this area of review and the associated acceptance criteria, which are summarized below, as well as the review interfaces with other SRP sections:

- In 10 CFR Part 50, Appendix A, 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 10 CFR Part 50, Appendix A, 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.

- In 10 CFR 52.47(b)(1), the NRC requires a DCA to contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC has been constructed and will be operated in accordance with the DC, the provisions of the AEA, and the NRC’s rules and regulations.

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.

The guidance in SRP Section 3.5.1.4 states that compliance with GDC 2 and GDC 4, on missiles generated by extreme winds, can be achieved by meeting the guidance in RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1, issued March 2007, and RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” Revision 0, issued October 2011.

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 230 mph.
  – A hurricane has a maximum 3-second gust of 290 mph.

- tornado-generated missile spectra

- A massive high-kinetic-energy missile that deforms on impact, such as a 4,000-pound automobile with dimensions of 16.4 feet by 6.6 feet by 4.3 feet, has a horizontal velocity of 135 feet per second (ft/s) and a vertical velocity of 91 ft/s.
- A rigid missile that tests penetration resistance, such as a 15-foot-long, 287-pound, 6-inch-diameter Schedule 40 pipe, has a horizontal velocity of 135 ft/s and a vertical velocity of 91 ft/s.
- A small rigid missile of a size that is sufficient to pass through openings in protective barriers, such as a 0.147-pound, 1-inch-diameter solid steel sphere, has a horizontal velocity of 26 ft/s and a vertical velocity of 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 307 ft/s.
- A rigid missile that tests penetration resistance, such as the Schedule 40 pipe described above, has a horizontal velocity of 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 225 ft/s.
- The design-basis vertical missile velocity for all missiles is 85 ft/s.

The applicant has assumed that the automobile missiles will impact at all altitudes of less than 30 feet above plant grade levels within 0.5 mile of the plant structures. However, DCA Part 2, Tier 2, Section 3.5.2, states that “the walls, roofs, and openings [of the RXB and CRB] are designed to withstand the design basis missiles discussed in Section 3.5.1.4.” It was unclear to the staff whether the entire height of the walls and the roof are designed to withstand the design-basis automobile missile or only the pipe and sphere missile. Therefore, the staff issued **RAI 8780**, **Question 03.05.01.04-1** (ADAMS Accession No. ML17188A455), requesting the applicant to clarify the above statements and to explain whether the automobile missile is applied to all evaluations or only up to 30 feet. In its response dated July 7, 2017 (ADAMS Accession No. ML17188A455) to **RAI 8780**, **Question 03.05.01.04-1**, the applicant stated that the portions of the RXB and CRB that are above the 30-foot plant elevation have not been analyzed to withstand the design-basis automobile missile but that they are resistant to the other design-basis missiles. The applicant also stated that it will revise DCA Part 2, Tier 2, Section 3.5.2, accordingly.

RAI 8780, **Question 03.05.01.04-1**, also specifies that, if the automobile missile is applied only to elevations of the RXB and CRB of less than 30 feet, the COL applicant needs to address the potential of automobile missiles striking above elevation 30′ because of elevated (above plant grade) parking lots. In its response to **RAI 8780**, **Question 03.05.01.04-1**, the applicant stated that it will add a COL information item (COL Item 3.5-3) to DCA Part 2, Tier 2, Section 3.5.2 and Table 1.8-2, to require a COL applicant to confirm that automobile missiles cannot be generated within a 0.5-mile radius of safety-related and risk-significant SSCs requiring missile protection that would lead to an impact higher than 30 feet above plant grade.

The staff reviewed the applicant’s response to **RAI 8780**, **Question 03.05.01.04-1**, and finds it acceptable because applying the automobile missile only to elevations below 30 feet is consistent with the guidance of RG 1.76 and RG 1.221 and because the applicant will revise DCA Part 2, Tier 2, Section 3.5.2, to be consistent with the assumptions of DCA Part 2, Tier 2, Section 3.5.1.4. In addition, the staff finds the proposed COL information item acceptable because it addresses the potential for an automobile missile impact higher than 30 feet. **RAI 8780**, **Question 03.05.01.04-1**, is resolved and closed, based on the applicant's incorporation of COL Item 3.5-3 into DCA Part 2, Tier 2, Section 3.5.2 and Table 1.8-2.

The guidance of RG 1.76 only applies 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 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 Inspections, Tests, Analyses, and Acceptance Criteria

Although there are no direct ITAAC for this section, DCA Part 2, Tier 1, Table 5.0-1, provides the parameters for design-basis tornado and hurricane winds and associated missile spectra. In addition, DCA Part 2, Tier 1, Section 3.11 and Section 3.13, "Control Building," specify that these design-basis loads are to be applied to the design of the RXB and CRB, respectively.

Based on its review, the staff finds the design descriptions cited above acceptable because they conform to the guidance in RG 1.76 and RG 1.221 for design-basis wind borne missiles for nuclear power plants. The NRC staff's evaluation of these two ITAAC items are addressed in SER Section 14.3. 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, missiles generated by tornadoes and extreme winds.

Table 3.5.1.4-1 NuScale COL Information Item 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 0.5 mile radius of safety-related SSCs and risk significant SSCs requiring missile protection that would lead to impact higher than 30 feet above plant grade. Additionally, if automobile missiles impact at higher than 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

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 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: No Tier 1 information is required for Section 3.5.1.5

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.5.1.5, addresses the need for the COL applicant that references the NuScale DCA to evaluate the potential for site proximity explosions and missiles and states that the COL applicant that references the NuScale DCA will provide site-specific information and evaluations in accordance with COL Item 2.2-1 and COL Item 3.5-2.

ITAAC: There are no ITAAC associated with Section 3.5.1.5.

Technical Specifications: There are no Technical Specifications for Section 3.5.1.5.

Technical Reports: There are no TRs for Section 3.5.1.5.

3.5.1.5.3 Regulatory Basis

10 CFR 52.47(a)(1) as a part of final safety analysis report (FSAR) must include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters

As a part of 10 CFR 52.47 (a)(3)(i) it is required to include the principal design criteria for the facility. Appendix A to CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units

In addition to 10 CFR 52.47(a)(1) and (a)(3)(i), the applicable regulatory requirements for identifying evaluation of 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

- 10 CFR Part 50, Appendix A, 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 NUREG-0800, provides a means for the applicant to meet the relevant requirements:

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

3.5.1.5.4 Technical Evaluation

Because the information on site proximity missiles (except aircraft) near the site is site specific, DCA Part 2, Tier 2, states that a COL applicant referencing the NuScale DCA will address the site-specific information on the evaluation of the potential for site proximity explosions and missiles caused by trains, trucks, ship or barge explosions, industrial facilities, pipeline explosions, or military facilities and others, in accordance with COL Item 2.2-1 and COL Item 3.5-2 in DCA Part 2, Tier 2, Table 1.8-2. If the total probability of an explosion is greater than an order of magnitude of $1x10^{-7}$ per year, 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.

3.5.1.5.5 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.

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: No Tier 1 information associated with Section 3.5.1.6.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.5.1.6, addresses the need for the evaluation of potential aircraft hazards and an aircraft hazard analysis and states that the COL applicant that references the NuScale DCA will provide site-specific information and evaluations in accordance with the guidance in RG 1.206 in accordance with COL Item 2.2-1.

ITAAC: There are no ITAAC associated with Section 3.5.1.6.

Technical Specifications: There are no Technical Specifications for Section 3.5.1.6.

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

3.5.1.6.3 Regulatory Basis

10 CFR 52.47(a)(1) as a part of final safety analysis report (FSAR) must include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters

As a part of 10 CFR 52.47 (a)(3)(i) it is required to include the principal design criteria for the facility. Appendix A to CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units

In addition to 10 CFR 52.47(a)(1) and (a)(3)(i), 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, GDC 4, as it requires SSCs important to safety 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 NUREG-0800, provides a means for the applicant to meet the relevant requirements:

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

3.5.1.6.4 Technical Evaluation

Because the information on potential aircraft hazards near the site is site specific, the applicant stated that the COL applicant that references NuScale DCA will demonstrate that the probability of aircraft hazards impacting the NuScale standard plant and causing consequences greater than the exposure guidelines in 10 CFR 100.21(c)(2) (which references 10 CFR 50.34(a)(1)) is less than $1x10^{-7}$ per year based on the COL applicant’s use of site-specific information in accordance with RG 1.206, “Applications for Nuclear Power Plants,” Revision 1, issued October 2018 (SRP Section 3.5.1.6, “Aircraft Hazards”), in accordance with COL Item 2.2-1 in DCA Part 2, Tier 2, Table 1.8-2.

3.5.1.6.5 Conclusion

As described above, the applicant has stated, in DCA Part 2, Tier 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.

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 states that to satisfy GDC 2 and GDC 4, safety-related 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 specify that design-basis loads, including those from extreme wind missiles, are to be applied to the design of RXB and CRB, respectively.

Technical Specifications: There are no TS 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

SRP Section 3.5.2, "Structures, Systems, and Components To Be Protected from Externally-Generated Missiles," Revision 3, issued March 2007, provides the relevant regulatory requirements for this area of review and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections:

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

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.

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

- RG 1.13, “Spent Fuel Storage Facility Design Basis,” Revision 2, 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

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. DCA Part 2, Tier 2, Section 3.5, states the following:

> *Safety-related SSC and those risk-significant SSC that have a safety function that would be relied upon following the missile producing event are potential missile targets. These SSCs are located inside the RXB and CRB. Table 3.2-1 lists SSC, their safety classification, and their risk significance.*

However, it is unclear to the staff which of the nonsafety-related, risk-significant SSCs listed in Table 3.2-1, if any, are required to be protected against externally generated missiles. Therefore, the staff issued **RAI 8804**, **Question 03.05.02-1** (ADAMS Accession No. ML17135A405), requesting the applicant to clarify which risk-significant SSCs have a safety function that would be relied upon following the missile-producing event and, thus, are required to be protected against externally generated missiles. In its response dated July 7, 2017 (ADAMS Accession No. ML17188A458), to **RAI 8804**, **Question 03.05.02-1**, the applicant stated that the SSCs that have a safety function that is relied upon following the missile-producing event and that, therefore, are required to be protected against externally generated missiles are listed in Table 3.2-1 and are identified as A1 (Safety-Related, Risk-Significant), A2 (Safety-Related, Nonrisk-Significant), and B1 (Nonsafety-Related, Risk-Significant). The staff finds the applicant’s response to **RAI 8804**, **Question 03.05.02-1**, acceptable because it clarifies that, for the NuScale design, all SSCs classified as A1, A2, and B1 will be protected against external missiles. Therefore, **RAI 8804**, **Question 03.05.02-1**, is considered resolved and closed.

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 seismic Category I structures. 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.

With respect to the RXB and CPB turbine missile protection, it 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 Inspections, Tests, Analyses, and Acceptance Criteria

Although there are no direct ITAAC for this section, DCA Part 2, Tier 1, Sections 3.11 and 3.13, specify that design-basis loads, including those from extreme wind missiles, are to be applied to the design of RXB and CRB, respectively.

Based on its review, the staff finds the design description cited above acceptable because it is in conformance with the guidance in RG 1.13 and RG 1.117 and, therefore, complies with the requirements of 10 CFR Part 50, Appendix A, GDC 2 and GDC 4.The staff evaluates the ITAAC in SER Section 14.3.7.4.4.

3.5.2.6 Combined License Information Items

The staff reviewed DCA Part 2, Tier 2, Section 3.5 and Table 1.8-2, for the appropriateness of identified COL action items as discussed in SRP Section 3.5.2, Review Procedure 4, pertaining to externally generated missiles. However, the staff did not identify a standard COL information item requiring a COL applicant to evaluate the potential for site-specific external hazards to produce a more energetic missile than the design-basis missiles. Therefore, the staff issued **RAI 8804**, **Question 03.05.02-2** (ADAMS Accession No. ML17135A405), requesting the applicant to include a COL information item in DCA Part 2, Tier 2, that requires a COL applicant that references the NuScale DC to evaluate site-specific hazards for external events that may produce more energetic missiles than the design-basis missiles defined in DCA Part 2, Tier 2, Section 3.5.1.4. In its response dated July 7, 2017 (ADAMS Accession No. ML17188A458), to **RAI 8804**, **Question 03.05.02-2**, the applicant stated that it will add a COL information item (COL Item 3.5-4) to DCA Part 2, Tier 2, Section 3.5.2 and Table 1.8-2. The staff finds the proposed COL information item (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. Therefore, **RAI 8804**, **Question 03.05.02-2**, is resolved and closed based on the incorporation of the COL information item into DCA Part 2, Tier 2, Section 3.5.2 and Table 1.8-2.

SER Table 3.5.2.6-1 lists the COL information item numbers and descriptions related to external missiles from DCA Part 2, Tier 2, Table 1.8-2.

Table 3.5.2.6-1 NuScale COL Information Items for Section 3.5.2

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 0.5 mile radius of safety-related SSCs and risk-significant SSCs requiring missile protection, that would lead to impact higher than 30 feet above plant grade. Additionally, if automobile missiles	3.5.2

Item No.	Description	DCA Part 2, Tier 2, Section
	impact at higher than 30 feet above plant grade, the COL applicant will evaluate and show that the missiles will not compromise safety-related and risk-significant SSCs.	
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 DCA Part 2, Tier 2, Section 3.5.1.4.	3.5.2

The staff finds the list in SER Table 3.5.2.6-1 to be complete. In addition, the list adequately describes actions necessary for the COL applicant. DCA Part 2, Tier 2, Table 1.8-2, does not need to include any additional COL information items for externally generated missile considerations.

3.5.2.7 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, Revision 0, Section 3.5.3, "Barrier Design Procedures," following the guidance in SRP Section 3.5.3, "Barrier Design Procedures," Revision 3, 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 missile impact. 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 of missile parameters are within the corresponding site parameters of the NuScale design. The COL applicant should reevaluate the SSCs important to safety in the NuScale design if the site characteristics of missiles are not within the corresponding site parameters of the NuScale design.

3.5.3.2 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: The applicant provided the barrier design procedures used for the NuScale design in DCA Part 2, Tier 2, Revision 0, Section 3.5.3. For the prediction of local damage from 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.

With regard to the overall damage predicted for a structure or barrier from tornado and hurricane missile impact, the applicant used Electric Power Research Institute (EPRI) NP440, “Full Scale Tornado Missile Impact Tests,” 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, “Design of Structures for Missile Impact,” Revision 2, 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.

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

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

3.5.3.3 Regulatory Basis

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

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

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

SRP Section 3.5.3, Revision 3, lists the acceptance criteria adequate to meet the above requirements and provides review interfaces with other SRP sections. In addition, the following guidance document provides acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments,” Revision 2, issued November 2011

3.5.3.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 2, Revision 0, 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 missile impact.

Local Damage Predication

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, Revision 0, Table 3.5-1, "Concrete Thickness to Preclude Missile Penetration, Perforation, or Scabbing." The staff reviewed DCA Part 2, Tier 2, Revision 0, Table 3.5-1, and finds that it only lists the concrete thickness to preclude perforation or scabbing from the design-basis hurricane and tornado pipe and sphere missiles. It does not include the massive high-kinetic-energy missile, such as an automobile, which is one of the missile threats to the barrier design. Therefore, the staff issued **RAI 8983**, **Question 03.05.3-1** (ADAMS Accession No. ML17216A837), to address this item. In its response dated September 25, 2017 (ADAMS Accession No. ML17268A251), to **RAI 8991**, **Question 03.05.03-1**, 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. The applicant stated that the "massive missile," such as the automobile missile, normally is very soft, and can be subjected to large deformations upon the impact. Therefore, for large structures, the dynamic response of the overall structure to this missile strike is negligible; the missile would be crushed upon impact and will not cause any significant local damage. Based on this review, the staff considers the RAI response acceptable because the 5 feet thick concrete barrier with 7000 psi compressive strength is relatively rigid compared to the automobile frame with a large foot print. The automobile impact load is distributed over a large footprint area on the barrier thus reducing the density of the applied impact load. Therefore, the automobile frame will tend to buckle and absorb the impact energy. Therefore, **RAI 8991**, **Question 03.05.03-1**, is resolved and closed.

For the concrete missile barriers, DCA Part 2, Tier 2, Revision 0, Section 3.5.3.1.1, "Concrete Barriers," provides the applicant's barrier design procedures. The applicant established penetration and spalling, perforation, and scabbing equations using the Modified National Defense Research Committee'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, Revision 0, Section 3.5.3.1.1, and finds that the applicant applied the Modified National Defense Research Committee'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 predication of local damage in the impacted area to be acceptable.

Based on its response dated September 25, 2017 (ADAMS Accession No. ML17268A251), to **RAI 8983**, **Question 03.05.03-2**, the applicant stated, in DCA Part 2, Tier 2, Revision 0, Section 3.5.3.1.2, "Steel Barriers," that the design does not use any steel missile barriers.

In DCA Part 2, Tier 2, Revision 0, Section 3.5.3.1.3, "Composite Barriers," the applicant stated that the design does not use composite barriers.

Overall Damage Predication

The staff reviewed DCA Part 2, Tier 2, Revision 0, Section 3.5.3.2, "Overall Damage Predication," and did not find the analysis procedure to address the design-basis sphere missile. Therefore, the staff issued **RAI 8983**, **Question 03.05.03-3** (ADAMS Accession No. ML17216A837), requesting the applicant to provide the missing information. In its response dated September 25, 2017 (ADAMS Accession No. ML17268A251), to **RAI 8983**, **Question 03.05.03-3**, 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 negligible to the overall structural response of the RXB and CRB.

In DCA Part 2, Tier 2, Revision 0, 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 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, Revision 0, 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. Therefore, the staff issued **RAI 8983**, **Question 03.05.03-3** (ADAMS Accession No. ML17216A837), requesting the applicant to provide justification of how the proposed alternative provides an acceptable method for complying with the relevant NRC acceptance criteria. In its response to **RAI 8991**, **Question 03.05.03-3**, dated September 25, 2017 (ADAMS Accession No. ML17268A251), 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 demonstrated that missile impact has virtually no effect on the overall response of 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 comply with 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.

Turbine Missile Barrier Design

On June 25, 2018, the applicant submitted revised DCA Part 2 sections and supporting information to the NRC staff for review. On July 16, 2018, the NRC conducted a public meeting to discuss the applicant's revised methodology for protection against a turbine missile event. During this public meeting, the staff discussed several questions and concerns and requested the applicant to make its supporting documentation available to address the questions. The staff issued **RAI 9596**, **Question 03.05.03-4** (ADAMS Accession No. ML18250A323). In its response dated October 31, 2018 (ADAMS Accession No. ML18304A306), to **RAI 9596**, **Question 03.05.03-4**, the applicant did not provide sufficient information to enable the staff to review the turbine missile barriers design, which is credited for providing essential SSCs contained within the barriers adequate protection against turbine-generated missiles. Therefore, an audit is required to review primarily nondocketed information and pertinent technical information to examine and verify the analyses and calculations; engineering drawings; design assumptions and the technical bases, including turbine missile size, mass, velocity; and related documentation that supports the fact that, consistent with DCA Part 2, Tier

2, Revision 2, Sections 3.5.1.3 and 3.5.3, the credited structural missile barriers (1) are designed to a bounding spectrum of turbine missiles, (2) have sufficient thickness to prevent penetration and spalling, perforation, and scabbing that could challenge the safety-related SSC, and (3) 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.

On March 19, 2019, April 16, 2019, and April 26, 2019, the NRC conducted public meetings to (i) discuss the barrier structural design issues regarding the analytical procedures and computer program used by the applicant for the local and overall structural damage prediction, and (ii) identify the information that the staff require to make its safety findings regarding the adequacy of RXB and CRB barrier structural design for the impact of design basis turbine missiles. On May 3, 2019, the applicant submitted a closure plan (ADAMS Accession No. ML19123A321) for the resolution of the staff identified design safety issues. The staff reviewed the applicant's proposed closure plan to resolve the missile barrier design issues and finds that the closure plan provides a path forward to resolve the missile barrier design safety issues. The staff is tracking resolution of turbine missile barrier design issue as **Open Item 03.05.03-1**.

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 regards to the procedures used for determining the effects and loadings on the RXB and CRB wall barriers from the impact of turbine missiles, the applicant submitted a closure plan (ML19123A321) to resolve the turbine missile barrier design issues. The staff will review and evaluate the complete and necessary information upon its submittal that is committed to in the closure the reactor and control building concrete walls provide sufficient protection to essential SSCs from the impact of design basis turbine missiles. The staff is tracking the resolution of turbine missile barrier design issues as **Open Item 03.05.03-1**.Based on its review of the closure plan, the staff finds that the closure plan provides a path forward to resolve the missile barrier design issues. ADAMS Accession No.

3.6 <u>Protection against Dynamic Effects Associated with the Postulated Rupture of Piping</u>

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 No. 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.

Technical Specifications: There are no TS 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, “Pipe Rupture Hazards Analysis,” Revision 0, issued September 2018, 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. In its letter dated December 20, 2018 (ADAMS Accession No. ML18354B400), the applicant submitted TR-0818-61384-P, Revision 1 dated December 20, 2018. The staff is tracking the applicant’s update of DCA Part 2, Tier 2, Section 3.6.5, which references TR-0818-61384-P, Revision 1, as **Confirmatory Item 03.06.01-1**.

3.6.1.3 Regulatory Basis

SRP Section 3.6.1, “Plant Design for Protection against Postulated Piping Failures in Fluid Systems outside Containment,” Revision 3, issued March 2007, provides the relevant regulatory requirements for this area of review and the associated acceptance criteria, which are summarized below, as well as review interfaces with other SRP Sections:

- 10 CFR Part 50, Appendix A, GDC 2, as it requires the protection of SSCs important to safety to withstand the effects of natural phenomena, such as earthquakes

- 10 CFR Part 50, Appendix A, 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

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.

DCA Part 2, Tier 2, Table 3.6-1, "High- and Moderate-Energy Fluid System Piping," identifies 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 19.3 kilograms per square centimeter (kg/cm^2) (275 psig)

The applicant defines moderate-energy system as a high-energy system that only operates 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 19.3 kg/cm^2 (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, "Protection against Postulated Piping Failures in Fluid Systems outside Containment," Revision 3, issued March 2007, and its companion BTP 3-4, "Postulated Rupture Locations in Fluid System Piping inside and outside Containment," Revision 2, 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 STP 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
- chemical and volume control system (CVCS)
- control rod assembly (CRA) and the control rod drive system (CRDS)
- containment system (CNTS)
- DHRS
- emergency core cooling system (ECCS)
- UHS

DCA Part 2, Tier 2, Table 3.6-1, identifies 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.

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. Section 3.6.1.4.2 and 3.6.2 of the Staff's SE evaluate the adequacy of the protection of essential SSCs form 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 and the CNV that require protection. The protection measure credited in the CNV are discussed in SER 3.6.2. The protection measures for the RXB are discussed further below (SER 3.6.1.3.2). 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 relative 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, "Fluid System Piping in Containment Penetration Areas," Section B.A.(ii).

- 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 LBB.

- For high- and moderate-energy systems that cannot be fully excluded using 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 BA.(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 postulated breaks is an acceptable methodology. In SER Sections 3.6.2 and 3.6.3, the staff evaluates the applicant's implementation of the pipe break location methodology and leak-before-break (LBB) methodology, respectively.

Additionally, the applicant stated that high-energy systems are analyzed for postulated circumferential breaks in fluid system piping greater than 1-inch nominal pipe size (NPS), longitudinal breaks in fluid system piping that is 4-inch NPS and greater, and leakage cracks in fluid system piping greater than 1-inch NPS. 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 postulating 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, Revision 0 (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. In its letter dated December 20, 2018, the applicant submitted TR-0818-61384-P, Revision 1. The staff is tracking the applicant's update of the reference in DCA, Part 2, Tier 2, Section 3.6.5, to TR-0818-61384-P, Revision 1, as **Confirmatory Item 03.06.01-1**.

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)

DCA Part 2, Tier 2, Section 3.6.1.1.2, identifies the main steam, 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 (CES) include moderate-energy lines in this area.

In its response dated December 4, 2018, (ADAMS Accession No. ML18338A238) to **RAI 8855**, **Question 03.06.02-13**, the applicant proposed changes to DCA Part 2, Tier 2, Section 3.6.2.1.2.2, "Break Exclusion." In DCA Part 2, Tier 2, Section 3.6.2.1.2.2, 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.

Because no high-energy failures have been postulated in the area outside the CNV (under the bioshield), the essential SSCs identified by the applicant do not require protection from the dynamic consequences of a high-energy pipe failure. In SER Section 3.6.2 the staff evaluates the break location criteria and the applicant's response to **RAI 8855**, **Question 03.06.02-13**. The staff is tracking the inclusion of the applicant's proposed changes into the next revision of the DCA as **Confirmatory Item 03.06.02-2.**

However, in BTP 3-3, Section B, Item 1.a(1), the staff indicates that, even though portions of the main steam 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 1 ft^2. This failure is postulated to establish the environmental conditions that the essential SSCs need to be protected (or designed) against.

In its response dated December 4, 2018, (ADAMS Accession No. ML148338A238) to **RAI 8855**, **Question 03.06.02-13**, the applicant proposed adding Section 3.6.2.1.2.1, "Non-mechanistic Secondary Line Breaks in Containment Penetration Area" to DCA Part 2, Tier 2. In the added section, the applicant stated that the 1-ft^2 flow area is disproportionately large for the NuScale SMR design. In the case of NuScale, the applicant pointed out that the minimum flow area described in BTP 3-3 exceeds the area of a full-circumferential rupture of the MSS piping (12-inch NPS) and the feedwater system (FWS) (4-inch NPS). The applicant proposed to postulate a nonmechanistic break of 12 square inches ($in.^2$) for the MSS and 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 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 12 $in.^2$ for the MSS and 5.87 $in.^2$ for the FWS.

Section 3.5.2.2 of TR-0818-61384-P, Revision 1, 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 1, 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, therefore, 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 a main steam 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 both, 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 a FWS break causes a subsequent break in an adjacent module.

The staff reviewed the applicant's approach of performing a bounding pipe break 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 address the as-design 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 only safety-related SSCs in the RXB (outside the bioshield) are the CIV hydraulic power units (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.

The staff evaluates the flooding consequences of pipe failures in SER Section 3.4.

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 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. 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 Inspections, Tests, Analyses, and Acceptance Criteria

In DCA Part 2, Tier 1, Section 2.1, the applicant identified a design commitment to ensure that safety-related SSCs are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems. DCA Part 2, Tier 1, Table 2.1-4, Item 4, requires the performance of an inspection of the as-built high- and moderate-energy piping systems and protective features for safety-related SSCs to ensure that they are installed in accordance with the as-built pipe break hazard analysis report and that safety-related SSCs are protected against, or qualified to withstand, the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.

The staff reviewed DCA Part 2, Tier 1, Table 2.1-4, Item 4, and finds that the proposed ITAAC are adequate to ensure that safety-related SSCs associated with the NPMs are protected against, or qualified to withstand, the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.

The staff evaluates the ITAAC in SER Section 14.3.3.

3.6.1.6 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

COL 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 No.	Description	DCA Part 2, Tier 2 Section
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 finds that DCA Part 2, Tier 2, Table 1.8-2, does need to include additional COL information items in relation to the PRHA.

3.6.1.7 Conclusion

Based on the discussion above and pending resolution of Confirmatory Items 03.06.01-1 and 03.06.02-2, 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 Introduction 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 that included using design provisions of separation, LBB, break exclusion, and protection devices against postulated high-energy piping ruptures in DCA Part 2, Tier 2, Section 3.6, "Protection against Dynamic Effects Associated with Postulated Rupture of Piping." In DCA Part 2, Tier 2, Revision 0, Section 3.6, the applicant initially stated that integral shield restraints (ISRs) are to be used to mitigate the dynamic effects of postulated HELBs. Traditional protection methods (pipe whip restraint and jet shielding) are needed when ISRs are not used to mitigate the dynamic effects of postulated HELBs. However, the applicant later determined that the NuScale plant design will no longer use ISRs and revised Section 3.6 in DCA Part 2, Tier 2, Revision 2, to address its alternate approach to mitigate the dynamic effects of postulated HELBs. In addition, in a letter dated December 20, 2018, the applicant submitted TR-0818-61384-P, Revision 1 (ADAMS Accession No. ML18354B400), to supplement the PRHA-related information contained in DCA Part 2, Tier 2, Section 3.6.2. Specifically, TR-0818-61384-P, Revision 1, 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 and in 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. DCA Part 2, Tier 2, Section 3.6.1.1, "Identification of High- and Moderate-Energy Piping Systems," and DCA Part 2, Tier 2, 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 Code Class 1, 2, or 3 criteria or the criteria in ASME B31.1.. In addition, 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. Moreover, 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. Moreover, 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. Furthermore, 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 DHRS lines and the bolted connection of reactor vent valves 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.

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

Technical Reports: The applicant identified TR-0818-61384-P as the relevant TR.

3.6.2.3 Regulatory Basis

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

- Compliance with 10 CFR Part 50, Appendix A, 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 staff issued several RAIs to the applicant to resolve its questions on the information in the original DCA submittal. As indicated in SER Section 3.6.2.2, the applicant initiated a rewrite of DCA Part 2, Tier 2, Section 3.6, to describe its alternate approach to mitigate the dynamic effects of postulated HELBs. The update to DCA Part 2 also includes an additional technical discussion reflecting the completion of TR-0818-61384-P, Revision 1, and some of the applicant’s responses to the staff’s RAIs on DCA Part 2, Tier 2, Section 3.6.2. This SER section focuses on the revised DCA and its compliance with the applicable NRC regulations and guidance instead of discussing each RAI and the applicant’s responses. The SER sections below discuss the staff’s review of DCA Part 2, Tier 2, Revision 2, Section 3.6, and TR-0818-61384-P, Revision 1.

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

DCA Part 2, Tier 2, Sections 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). 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 in the sections below.

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. 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 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 steam lines attach. In addition, outboard of the valves in each of these lines is a short piping segment welded to a flange used to connect the refueling pipe spools to the NPM module.

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 also 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 FW piping from containment to the first isolation valve. Similarly, in DCA Part, Tier 2, Section 3.6.2.7, the applicant described how the break exclusion criteria are applied to the segments of piping between the main steam and FW lines from containment to the penetration at the reactor pool wall (including tees to the DHRS) and the DHRS piping outside containment.

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.

Based on its review of DCA Part 2, Tier 2, Sections 3.6.2.1.2 and 3.6.2.7, the staff determined that the applicant had not adequately justified its application of the break exclusion in the areas described above. In certain cases, the applicant expanded the break exclusion area beyond those portions of piping in containment penetration areas as delineated in BTP 3-4, Section B, Item A(ii). In addition, it was not clear that some of the applicant's design provisions and the weld inspection requirement in the break exclusion area, as currently described in DCA Part 2, are consistent with the pertinent staff guidance in BTP 3-4, Section B, Item A(ii)(7). The staff notes that DCA Part 2, Tier 2, Sections 3.6.2.1.2 and 3.6.2.7, primarily address the NuScale design and inspection requirement for system piping within the break exclusion area. To support the staff's safety determination on the acceptability of the NuScale break exclusion areas identified in DCA Part 2, Tier 2, Sections 3.6.2.1.2 and 3.6.2.7, the applicant should provide certain 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, Sections 3.6.2.1.2 and 3.6.2.7, are considered and applied to the results of the design of these portions of the system piping, including any associated welds. Therefore, the staff issued **RAI 8836**, **Question 03.06.02-2** (ADAMS Accession No. ML17153A065), requesting the applicant to provide the additional information to justify its application of the break exclusion in the areas identified in DCA Part 2, Tier 2, Sections 3.6.2.1.2 and 3.6.2.7.

In its responses dated January 25, 2019 (ADAMS Accession No. ML19025A280), and February 6, 2019 (ADAMS Accession No. ML19037A319), to **RAI 8836**, **Question 03.06.02-2**, related to its break exclusion criteria and the application of these criteria to the NuScale plant design, the applicant stated that, where piping connects to a CNV safe-end, as illustrated in Figures 1 through 4 of its response, 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 also 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), to ensure a low probability of rupture. DCA Part 2, Tier 2, Section 6.6, “Inservice Inspection and Testing of Class 2 and 3 Systems and Components,” and Section 6.2, “Containment Systems,” detail the inservice examination requirements.

Moreover, in its response to **RAI 8836**, **Question 03.06.02-2**, the applicant stated that it evaluated the ASME BPV Code Class 1, 2, and 3 portions of the piping system, including their associated branch piping in the break exclusion area, to the relevant break exclusion stress and fatigue criteria as delineated in NuScale DCA Part 2, Tier 2, Section 3.6.2.1.2. In addition, the applicant provided a tabulated, quantitative summary of the calculated maximum stress ranges with a comparison of the applicable break exclusion allowable stress and fatigue limit for each of the applicable piping segments within the break exclusion areas. Moreover, the applicant referenced TR-0818-61384-P for 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. Furthermore, the applicant included its proposed markups of DCA Part 2, Tier 2, Table 6.2-3, “Containment Vessel Inspection Elements,” and Table 6.6-1, “Examination Categories and Methods,” and Appendix A, “Break Exclusion—Compliance with Regulatory Acceptance Criteria,” to TR-0818-61384-P.

In its response dated December 4, 2018 (ADAMS Accession No. ML18338A238), to **RAI 8855**, **Question 03.06.02-13** (ADAMS Accession No. ML17153A065), the applicant included a markup of DCA Part 2, Tier 2, Section 3.6.2.1.2.2; Section 3.6.2.1.2.3, “Leakage Cracks”; and Figure 3.6-33, “Application of BTP 3-4 Break Location Guidance in the NPM bay and RXB,” to clarify its application of the staff’s guidance in BTP 3-4, Section B, Items A(ii), A(iii), and A(v), for postulating break locations.

Based on the review of the above information in the applicant’s responses to **RAI 8836**, **Question 03.06.02-2**, and **RAI 8855**, **Question 03.06.02-13**, including the applicant’s proposed markups of DCA Part 2, Tier 2, Table 6.2-3, Table 6.6-1, Section 3.6.2.1.2.2, Section 3.6.2.1.2.3, and Figure 3.6-33, and of TR-0818-61384-P, Appendix A, the staff found that the applicant has provided sufficient information to explain how the NuScale break exclusion area design provisions are considered and applied to the design of the portions of system piping in the break exclusion area. For the piping segments for which the break exclusion criteria are applied, the results of the tabulated, quantitative summary of the calculated maximum stress ranges are within the relevant BTP 3-4 stress and fatigue limits for postulating break locations. Accordingly, the staff finds the applicant’s response acceptable because the applicant has adequately demonstrated its design provisions and specified a 100-percent volumetric inservice examination, which meet the applicable BTP 3-4 break exclusion criteria in the NRC’s guidelines, and because the applicant has appropriately justified the acceptability of NuScale’s application of the break exclusion area. The staff is tracking **RAI 8836**, **Question 03.06.02-2**, and **RAI 8855**, **Question 03.06.02-13**, as **Confirmatory Items 03.06.02-1 and 03.06.02-2**, respectively.

DCA Part 2, Tier 2, Section 3.6.2.7, states that each of three reactor vent valves (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. In its responses dated March 27, 2018 (ADAMS Accession No. ML18086B442), September 13, 2018 (ADAMS Accession No. ML18256A300), November 15, 2018 (ADAMS Accession No. ML18319A380), December 13, 2018 (ADAMS Accession No. ML18348A917), and January 22, 2019 (ADAMS Accession No. ML19022A364), to **RAI 9358**, **Question 03.06.02-17** (ADAMS Accession No. ML18026A519), the applicant provided its justification to ensure that the bolted connection provides confidence that the probability of gross rupture is extremely low and, therefore, is classified as a break exclusion area. The applicant stated that the components that comprise these bolted connections (valves, bolts, and nozzles) are classified as ASME Code Class 1 components and are designed, fabricated, constructed, tested, and inspected in accordance with the ASME Code, Section III, Subsection NB. The applicant also stated that the stress design criteria specified in ASME 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 they meet the intent of the guidance in BTP 3-4 for typical piping systems. In addition, the applicant discussed the conservatism included in the fatigue evaluation for the NuScale RVV and RRV bolted connection. Moreover, 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 then explained why the NuScale RVV and RRV bolted connections are not susceptible to these types of phenomena. Furthermore, 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. The NRC staff's found that the above 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 would provide confidence to ensure that the probability of gross rupture at the bolted connection is extremely low and the bolted connection would be considered as a break exclusion area. However, the applicant has not yet completed the stress and fatigue calculations to ensure that those stress and fatigue design criteria are met.

In a letter dated February 27, 2019 (ADAMS Accession No. ML 19058A670), theapplicant provided a supplemental response to address the update of the DCA with a summary of those key technical justifications for the break exclusion at the RVV/RRV bolted connections. In addition, in a letter dated February 4, 2019 (ADAMS Accession No. ML19035A682), the applicant indicated that it will complete the stress and fatigue evaluation for the RVV/RRV bolts in July 2019. The staff is tracking **RAI 9358**, **Question 03.06.02-17**, as **Open Item 03.06.02-1**.

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

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 cumulative usage factor exceeds 0.1, or 0.4 with consideration of environmentally assisted fatigue. 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 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.

Moreover, 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 DCA definition of a terminal end acceptable because it conforms to guidance in BTP 3-4, Footnote 3, for postulating pipe ruptures.

Furthermore, DCA Part 2, Tier 2, Section 3.6.2.1.8, states that, for high-energy lines, with the exception of 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 Code, Section III, Class 1, piping for which a stress analysis has been performed, leakage cracks are postulated at axial locations where the calculated stress range by Equation (10) in ASME BPV Code Section III, NB-3653, exceeds 1.2 S_m. For ASME 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.

Moreover, 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.

Furthermore, 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. Moreover, 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., finite element analysis). 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, Revision 2, Section 3.6.2.1.8, originally stated that the leakage cracks for high- and moderate-energy piping should be postulated to be in the circumferential locations that result in the most severe environmental consequences. In its response dated November 5, 2018 (ADAMS Accession No. ML18309A365), to **RAI 8836**, **Question 03.06.02-12** (ADAMS Accession No. ML17153A065), the applicant included a markup of DCA Part 2, Tier 2, Section 3.6.2.1.8, that states that the leakage cracks for the 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. Based on its review of the applicant's response and the markup to DCA Part 2, Tier 2, Section 3.6.2.1.8, the staff determined that the proposed DCA change 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. The staff is tracking **RAI 8836**, **Question 03.06.02-12**, as **Confirmatory Item 03.06.02-3**.

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. Moreover, it 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 1, 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 respective applicant’s dynamic analysis methodologies are described below.

<u>Blast Effects</u>

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 a 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 1, 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. Moreover, 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 1, 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. Moreover, 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 1 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. Moreover, 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 one-twenty-fifth of that for a typical large LWR.

Furthermore, 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, is acceptable. Specifically, the applicant has 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. Moreover, the CFD analysis was benchmarked against several experiments and analyses of similar conditions studied in the literature to verify its suitability. The applicant has 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 on the blast wave effects, as identified in SRP Section 3.6.2, Appendix A.

Jet Impingement Loads

DCA Part 2, Tier 2, Section 3.6.2.2.3, and TR-0818-61384-P, Revision 1, 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 a HELB yielding a single-phase steam jet, a HELB yielding a two-phase steam/water jet, and a 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, 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, 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 1, Appendix E, the applicant 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 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, 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. Moreover, 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, Table E-2, 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 1 inch 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. Furthermore, 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, 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. Moreover, the applicant stated that, if one of these criteria is not met, a resonance is implausible.

Furthermore, 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 asymmetry, 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. Moreover, 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 on the potential dynamic jet amplification and resonance jet impingement effects identified in SRP Section 3.6.2, Appendix A.

Pipe Whip Effects

DCA Part 2, Tier 2, Section 3.6.2.2.2, and TR-0818-61384-P, 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 distance 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 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. Moreover, 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, 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.

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. In its response dated November 5, 2018 (ADAMS Accession No. ML18309A364), to **RAI 8836**, **Question 03.06.02-3** with regard to the pipe whip restraints design, the applicant included a markup of DCA Part 2, Tier 2, Section 3.6.2.3.1.1, that describes the design criteria for the pipe whip restraints for 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. Moreover, 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 the NuScale's pipe whip restraints design criteria, as provided in the markup of 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. The staff is tracking **RAI 8836**, **Question 03.06.02-3**, as **Confirmatory Item 03.06.02-4**.

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. The staff did not find any information (or pointer) in DCA Part 2, Tier 2, Section 3.6, that addresses the initial condition assumed for evaluating the dynamic response of the postulated breaks. Therefore, the staff issued **RAI 8855**, **Question 03.06.02-14** (ADAMS Accession No. ML17153A065), requesting the applicant to clarify the piping system initial conditions assumed in the pipe motion and dynamic effects of the postulated breaks analysis and compare this with the staff guidance in SRP Section 3.6.2, Section III, Item 2.A. In its response dated January 31, 2019 (ADAMS Accession No. ML19031C976), to **RAI 8855**, **Question 03.06.02-14**, the applicant provided a markup of DCA Part 2, Tier 2, Table 3.6-4, "NuScale Power Module Piping Systems Design and Operating Parameters," which states 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. Based on its review of the applicant's response and the markup of DCA Part 2, Tier 2, Table 3.6-4, the staff determined that the proposed DCA change is consistent with the pertinent staff guidance in SRP Section 3.6.2, Section III, Item 2.A, and, therefore, is acceptable. The staff is tracking **RAI 8855**, **Question 03.06.02-14**, as **Confirmatory Item 03.06.02-5**.

3.6.2.4.2 *Pipe Rupture Hazards Analysis Report*

To support the staff's review and the safety determination on 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, Section 2.1, also 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 SMR. 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 2-inch NPS. 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.

Moreover, 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. Figure 3.1, "Flowchart of Methodology for Evaluation of Line Breaks," of TR-0818-61384-P, describes 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. In addition, Table 5-1, "Summary of Approach and Result for Line Break Assessment by Plant Area," of TR-0818-61384-P, summarizes 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. Moreover, 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 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 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.

However, the staff notes that the PRHA in TR-0818-61384-P has not been incorporated by reference into the NuScale DCA. In an e-mail dated January 23, 2019, the applicant stated that it will update DCA Part 2, Tier 2, Section 1.6, “Material Referenced,” Table 1.6-2, “NuScale Referenced Technical Reports,” to specify that TR-0818-61384-P is incorporated by reference into the NuScale DCA. In the same e-mail, the applicant committed to amend TR-0818-61384-P, where applicable, to specify that 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, for piping. Moreover, the applicant stated that it will revise TR-0818-61384-P, where applicable, because the blast effects load is small and, therefore, its effect on load combinations is inconsequential. Therefore, its exclusion does not affect compliance with the applicable ASME Code allowable limits. The applicant stated that it will address these items in a letter with revisions to TR-0818-61384-P, Revision 1. The staff is tracking the applicant’s commitment to amend TR-0818-61384-P for these PRHA-related items as **Open Item 03.06.02-2**.

In addition, to correct certain inconsistencies between the NuScale DCA and the TR-0818-61384-P the applicant has committed to clarify the discussion of essential SSCs in the RXB as addressed in DCA Part 2, Tier 2, Section 3.6.2.2.3 and to clarify the scope of the text discussion included in DCA Part 2, Tier 2, Section 3.6.2.2. The staff is tracking the applicant’s commitment to amend DCA Part 2, Tier 2, Sections 3.6.2.2 and 3.6.2.2.3, as **Open Item 03.06.02-3**.

Moreover, the staff noted that 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 of mechanical and electrical equipment, that are not within the review scope of SRP Section 3.6.2 and that, therefore, 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. Moreover, in a letter dated December 4, 2018 (ADAMS Accession No. ML18338A238), and in TR-0818-61384-P, 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.

Furthermore, the applicant’s PRHA in TR-0818-61384-P 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, Section 3.6.2.2.3, the final routing of piping, including placement of restraints beyond that 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.

3.6.2.4.3 Inspections, Tests, Analyses, and Acceptance Criteria

DCA Part 2, Tier 2, Section 14.3, discusses the bases, processes, and selection criteria used to develop Tier 1 information. 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 the protection of the safety-related SSCs of the NPM and RXB against postulated pipe rupture effects. The NRC staff’s evaluation of these two ITAAC items are addressed in SER Section 14.3.3.4.1.2 of this SER.

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

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.	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 an evaluation and disposition 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

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 those three COL information items adequately describe the respective actions necessary for COL applicants to complete with regard to PRHAs for their associated plant areas. Specifically, the staff finds them acceptable because they specify that the COL applicant that reference the NuScale Power plant design certification need to complete the routing of the applicable piping systems, to update the associated pipe rupture hazards analyses, and to evaluate multi-module impacts in common pipe galleries and subcompartment pressurization. The staff will confirm that the COL applicant provides this information with the COL application.

3.6.2.6 *Conclusion*

Given the open items described above, the staff is unable to reach a conclusion associated with this section of its review. Following resolution of these open items, the staff will determine whether the applicant’s descriptions of the PRHA for the NuScale Power Plant design comply with the applicable requirements in 10 CFR Part 50, Appendix A, GDC 4.

3.6.3 Leak-Before-Break Evaluation Procedures

3.6.3.1 Introduction

This section of DCA Part 2 describes the LBB evaluation procedures. As stated in GDC 4, dynamic effects associated with postulated pipe ruptures may be excluded from the design basis after the Commission reviews and approves the LBB analysis to demonstrate that the probability of a fluid system piping rupture is extremely low under conditions consistent with the design for the piping. The Commission must independently review and confirm a plant's ability to detect a leak in the piping components well before the onset of an unstable crack growth (i.e., structural failure).

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, "NuScale Power Module," 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 LBL evaluation procedures.

The applicant has 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. DCA Part 2, Tier 2, Figures 3.6-2, 3.6-3, and 3.6-5, show schematics of the two MSS lines and the two FWS lines that were analyzed for LBB.

The applicant has followed the 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 the LBB.

Technical Specifications: There are no TS specifically related to LBB materials and design.

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:

- 10 CFR Part 50, Appendix A, 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 Code.

LBB should only be applied to high energy, ASME 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 Code Classes 1 and2.

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 is not limited to those postulated pipe rupture locations determined from NUREG-0800, 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.

<u>Potential Degradation Mechanisms for Piping</u>

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

<u>Erosion/Corrosion</u>

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 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 Code requirements.

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 has 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 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 90 kilopounds per square inch. 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 that the MSS piping and FWS piping being evaluated for LBB are not susceptible to failure by SCC.

Creep and Creep Failure

Creep and creep failure are typically not of concern for austenitic stainless steel piping below 426.7 degree 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.

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.

<u>Fatigue</u>

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.

<u>Thermal Aging Embrittlement</u>

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 (FNs) of 5-20 and 5-16, respectively, and is consistent with the guidance in RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 4, 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, the staff concludes that thermal aging is not a concern in the piping systems that the applicant is qualifying for LBB.

<u>Thermal Stratification</u>

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 in not likely.

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.

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).

- 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.

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.

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:

- 8-inch NPS piping connecting perpendicularly to 12-inch NPS piping
- a transition where 12-inch NPS piping reduces to be welded to an 8-inch NPS 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 8-inch NPS pipes and the 12-inch NPS pipes and the transition where the 12-inch NPS piping reduces to be welded to the 8-inch NPS elbow. The applicant stated that, at these locations, the stress points were below the SBAC and that, 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 (IGSCC) 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 is currently reviewing the methodology and cracking mechanisms used as part of the confirmatory analysis. This is being tracked as **Confirmatory Item 03.06.3-1**

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 is currently reviewing the methodology and performing a confirmatory analysis of the information and data provided by the applicant. This is being tracked as **Confirmatory Item 03.06.3-1.**

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 February 8, 2018 (ADAMS Accession No. ML173198002), 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 8-inch NPS elbow. The applicant clarified that the crack analyzed for the 8-inch NPS elbow is the through-wall circumferential crack at extrados. The staff is currently reviewing the methodology and performing a confirmatory analysis of the information and data provided by the applicant. This is being tracked as **Confirmatory Item 03.06.3-1.**

<u>Leak Detection</u>

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 CES 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.

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, pending the results of the confirmatory analysis and resolution of the confirmatory items.

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

- 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.

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, "Site Parameters," 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 TS associated with DCA Part 2, Tier 2, Section 3.7.1.

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 10 CFR Part 50, Appendix A, 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 pertinent 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 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 documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- DSRS Section 3.7.1, Revision 0, 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, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” Revision 2, 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, “Damping Values for Seismic Design of Nuclear Power Plants,” Revision 1, 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, 1998 (ADAMS Accession No. ML081400293)

- DC/COL-ISG-017, “Interim Staff Guidance on 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. In accordance with DSRS Section 3.7.1, the staff evaluated DCA Part 2, Tier 2, Revision 2, Section 3.7.1, and DCA Part 2, Tier 1, Revision 2, Chapter 5, with regard to seismic design parameters. 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 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 the 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 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 established 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 the approach for the specification of the OBE and exclusion of the seismic analysis and design for the OBE is acceptable because it complies with the agency's regulatory requirements.

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. 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 equivalent 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.
- Additional control frequency points are established below 3.5 hertz, and the control points above 3.5 hertz are shifted to higher frequencies.
- The zero-period acceleration frequency increases from 33 hertz to 50 hertz.

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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 address the high-frequency hard rock sites that will also be used to evaluate seismic Category I structures for hard rock sites. The CSDRS-HF has a lower frequency content below approximately 10 hertz and higher frequency content above approximately 10 hertz than the CSDRS. The CSDRS-HF is applied at three mutually orthogonal directions—two horizontal and one vertical. 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 equivalent 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, on 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 six sets of time histories (five for the CSDRS and one for the CSDRS-HF), with each set consisting of three components (the two horizontal components for the E-W direction and N-S direction and the vertical component). The associated time histories were developed to envelop the CSDRS and the CSDRS-HF in conformance with the acceptance criteria in DSRS Section 3.7.1.II.1.B, Option 1, Approach 2, Revision 0. The sections below present the staff’s technical evaluation of the seed records and design ground motion time histories.

<u>Seed Records for Development of the Certified Seismic Design Response Spectra and the Certified Seismic Design Response Spectra-High Frequency Matched Time Histories</u>

The five sets of time histories used to match or envelop the CSDRS were based on the three ground motion components recorded from the magnitude 7.3 Landers, CA, earthquake (Yermo) event that occurred on June 28, 1992; the magnitude 6.9 Loma Prieta, CA, earthquake (Capitola) event that occurred on October 17, 1989; the magnitude 7.6 Chi-Chi, Taiwan, earthquake (Chi-Chi) event that occurred on September 21, 1999; the magnitude 7.4 Kocaeli, Turkey, earthquake (Izmit) event that occurred on August 17, 1999; and the magnitude 6.9 Imperial Valley, CA, earthquake (El Centro) event that occurred on May 18, 1940. The same magnitude 7.3 Landers, CA, earthquake that was recorded 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 0.005-second time interval to achieve a Nyquist frequency of 100 hertz.

<u>Meeting the Criteria in Design-Specific Review Standard Section 3.7.1, Revision 0, Option 1, Approach 2</u>

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 hertz to 100 hertz:

- 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 second, which is small enough to provide a Nyquist frequency of 100 hertz.

- 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 hertz.

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 bio shield, the fuel storage rack, and the reactor building crane.” This statement may imply that the applicant did not use all the sets of time histories for the design of all the SSCs. In **RAI 8900, Question 03.07.01-01**, the staff requested the applicant to address, in detail, any exceptions in the DCA and to include a basis for not using all the CSDRS and CSDRS-HF compatible time histories (and the resulting in-structure response spectra (ISRS)) for the design of all the SSCs. In its response dated September 5, 2017 (ADAMS Accession No. ML17249A965), the applicant stated that all of the associated 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 the Lucerne seed time histories. The applicant provided markup copies of two tables that show the definition of the eight seismic analysis identification codes and 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. However, based on the applicant's response, it appears to the staff that the seismic design basis of the seismic Category I system and components may not be the same as the seismic design basis of the seismic Category I structures. Because the applicant established both the CSDRS and CSDRS-HF as its standard site design parameters, it implies that the standard seismic design uses both spectra as input to the design of all the SSCs.

In a public meeting with the applicant on November 7, 2017, the staff requested the applicant to further justify why it did not use the seismic design requirement as prescribed in DCA Part 2, Tier 2, Section 3.7.1, and DCA Part 2, Tier 1, Section 5.0, for all the seismic Category I SSCs. DCA Part 2, Tier 1, and DCA Part 2, Tier 2, should include this justification and clarification on the use of different seismic design requirements for the qualification of systems and components. DCA Part 2 should also include Tables 1 and 2 of the applicant's response, which identify the seismic input used for the qualification of the SSCs within the scope of the NuScale standard design. The applicant indicated that it will provide its response to the staff's question as a supplemental response to **RAI 8935**, **Question 03.07.02-24**.

In its supplemental response dated December 7, 2017 (ADAMS Accession No. ML17341B655), to **RAI 8935**, **Question 03.07.02-24**, the applicant provided markup copies that show that (1) it has updated DCA Part 2, Tier 1, Table 5.0-1, and DCA Part 2, Tier 2, Table 2.0-1, to clarify 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 and (2) it added notes to 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, to clarify the basis of the design-basis seismic loads for applicable SSCs. The staff reviewed the applicant's supplemental response and accompanying markup copies and finds the response to be acceptable because it adequately addressed the staff's concern. The staff confirmed that the applicant revised DCA Part 2, Tier 2, Revision 2, dated October 30, 2018, as it had committed to do in its response to **RAI 8935**, **Question 03.07.02-24**. Accordingly, **RAI 8935**, **Question 03.07.02-24**, is resolved.

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.

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."

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 ISRS.

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.

In summary, the staff reviewed the applicant’s percentage of critical damping values used in its analyses of the seismic Category I and II SSCs and finds it to be acceptable because (1) the damping values are in accordance with RG 1.61, Revision 1, and (2) the prescription of percentage of critical damping values meets DSRS Section 3.7.1.II.2.

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 meters (86 feet) and 16.8 meters (55 feet) 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 meters per second (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 241.9 m/s (793.3 ft/s) to 2,439 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 2,439 m/s (8000 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 could 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

COL 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	3.7.1.1.3

COL Item No.	Description	DCA Part 2, Tier 2 Section
	and distance of the appropriate 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. 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 • 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

The staff reviewed the COL information items provided in revision 0 of the DCA Part 2, Tier 2, Section 3.7.1 and determined that additional COL information item may be necessary. 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. In **RAI 8900**, **Question 03.07.01-02**, the staff requested the applicant to include a COL information item that requires a COL applicant to verify and ensure the adequacy of the seismic input motion and the deterministic soil columns used in the site-specific SSI analysis of the deeply embedded seismic Category I structures.

In its response dated September 5, 2017 (ADAMS Accession No. ML17249A965), the applicant revised DCA Part 2, Tier 2, Section 3.7.1 and Table 1.8-2, to include a COL information item that requires a COL applicant that references the NuScale DC to ensure the adequacy of the seismic input motion and the deterministic soil columns used in the site-specific SSI analysis of its seismic Category I structures. Specifically, the applicant updated COL Item 3.7-2 to include additional information. The applicant stated that the COL applicant must perform the additional site-specific seismic analysis to confirm the adequacy of the seismic input motion and deterministic soil columns used in the SSI analysis. The FIRS are the starting points 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 are 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.

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 reviewed the information provided by the applicant in regard to the addition of a COL information item that verifies and ensures the adequacy of the seismic input motion and the deterministic soil columns used in the site-specific SSI analysis of the deeply embedded seismic Category I structures. The staff finds the information to be acceptable because the applicant (1) included a new 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 take 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 in accordance the regulatory guidance in DC/COL-ISG-017. The staff confirmed that the applicant revised DCA Part 2, Tier 2, Revision 2, dated October 30, 2018, as it had committed to do in its response to **RAI 8900**, **Question 03.07.01-02**. Accordingly, **RAI 8900**, **Question 03.07.01-02**, is resolved.

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, 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, (Revision 2 hereinafter, unless specified otherwise), 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, "Site Parameters," 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, addresses site-specific structures not within the scope of the NuScale standard design. DCA Part 2, Tier 1, Table 5.0-1, specifies site design parameters used in the NuScale standard design. DCA Part 2, Tier 1, Figures 5.0-1 and 5.0-2, specify the CSDRS for all seismic Category I SSCs, and DCA Part 2, Tier 1, 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 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.

Technical Specifications: There are no TS 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:

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

- 10 CFR Part 50, Appendix S, as it requires that, for 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, "Seismic System Design," Revision 0

- RG 1.61, Revision 1

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

- RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components," Revision 1, 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 analysis methods, analytical modeling for SSI effects, development of ISRS, combination of spatial and modal responses, the consideration of torsional effects, an 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 a 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. However, DCA Part 2, Tier 2, Revision 0, Section 3.7.2, did not contain information with respect to the seismic analysis methods applied to the SSCs. Therefore, in **RAI 9036**, **Question 03.07.02-27**, the staff requested the applicant to provide information on the seismic analysis methods that it used for the respective NuScale seismic Category I SSCs.

In its response dated October 17, 2017 (ADAMS Accession No. ML17290B261), to **RAI 9036**, **Question 03.07.02-27**, 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 (the 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 applicant's response and proposed DCA markups and finds 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. The staff confirmed that the applicant incorporated its proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Based on this review, **RAI 9036**, **Question 03.07.02-27**, is resolved and closed.

<u>Dry Dock Modeling</u>

DCA Part 2, Tier 2, Section 3.7.2.1.2.1, provides information on analysis methods for the RXB that includes pool water in the UHS. Specifically, the staff notes that the seismic analysis assumes that the dry dock is full of water and part of the UHS, and the nominal water level is set at elevation 94′. In DCA Part 2, Tier 2, Revision 0, Section 9.1.3, the staff notes that the dry dock can be drained partially or completely to support plant operations. In DCA Part 2, Tier 2, Section 9.1.3.3.5, the staff further notes 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 about 12 feet. 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. Therefore, in **RAI 8933**, **Question 03.07.02-16**, the staff requested the applicant to provide a technical basis for not considering different water level conditions for the dry dock in the seismic analysis. In addition, the staff requested the applicant to address the effect of potential variation in the UHS water level on the seismic analysis of the RXB and NPM. The staff also requested the applicant to describe the analysis and design criteria to ensure that no adverse seismic interaction occurs between the dry dock gate and adjacent seismic Category I SSCs.

In the response to **RAI 8933 Question 03.07.02-16** dated October 31, 2018 (ADAMS Accession No. ML18304A476) and, in the supplemental responses dated January 21, 2019 (ML19021A016) and February 11, 2019 (ADAMS Accession No. ML19042A646), the applicant indicated that a sensitivity study was performed to determine the effect of an empty dry dock on the response of the RXB. 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% structural damping for ISRS generation, and 7% 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 also identified information in the RAI response that the dry dock gate is designed to withstand the effect of the Safe Shutdown Earthquake 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 3–7, 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. In the proposed markups included in the RAI response, the applicant incorporated the summary of relevant portions of the RAI response into DCA Part 2 Tier 2 Section 3.7.2.

The staff finds the applicant's RAI response with its proposed DCA markups to be acceptable because the applicant 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. Based on this review, the staff is tracking **RAI 8933 Question 03.07.02-16,** as **Confirmatory Item 03.07.02-1,** pending the applicant's update to the next revision of the DCA.

<u>Fluid-Structure Interaction Correction Factor</u>

In DCA Part 2, Tier 2, Section 3.7.2.1.2.4, 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 addressees 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 that accounts for the FSI effects is established.

In DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.2.4, Equation 3.7-14 represents a conversion of the ANSYS FSI hydrodynamic pressure to the SASSI2010 equivalent static pressure. In this process, the ANSYS analysis used the CSDRS-compatible Capitola time history input on a fixed-base model, and the SASSI2010 analysis used the CSDRS-compatible Capitola time history input for soil types 7, 8, and 11, respectively. However, the staff notes that the applicant did not consider the CSDRS-HF compatible time history input for soil type 9 (hard rock) for FSI correction factors. Because the boundary conditions for an ANSYS fixed-base model and a SASSI model with soil type 9 (hard rock) are similar, it appears that FSI correction factors developed for soil type 9 may be more representative. Further, DCA Part 2, Tier 2, Section 3.7.2.1.2.4, describes a method used to account for the missing FSI effects by adding 1.28g vertical static loading to the SAP2000 model. However, the applicant did not provide sufficient information to enable the staff to evaluate the adequacy of the method used. Therefore, in **RAI 8934**, **Question 03.07.02-15**, the staff requested the applicant to (1) explain why it did not consider the FSI correction factors for the case of the CSDRS-HF compatible time history input for soil type 9 and (2) provide a technical basis for the method used to account for the missing FSI effects.

In its response dated December 21, 2017 (ADAMS Accession No. ML17355A678), and in its supplemental responses dated May 7, 2018 (ADAMS Accession No. ML18127B711); June 29, 2018 (ADAMS Accession No. ML18180A343); August 27, 2018 (ADAMS Accession No. ML18239A307); August 28, 2018 (ADAMS Accession No. ML18240A436); and October 19, 2018 (ADAMS Accession No. ML18292A747), to **RAI 8934**, **Question 03.07.02-15**, the applicant provided the following information:

- The applicant stated that the FSI correction factors based on the analysis cases of the CSDRS and soil types 7, 8, and 11 envelop the analysis case of the CSDRS-HF and soil type 9. To support this conclusion, the applicant provided information showing that the overall RXB base reactions computed from runs using soil types 7, 8, and 11 and CSDRS-compatible Capitola time history input envelop those from runs using soil types 7 and 9 and CSDRS-HF compatible Lucerne time history. The applicant also provided information indicating that the hydrodynamic pressures on the pool walls and foundation from the ANSYS dynamic analysis with the CSDRS and CSDRS-HF as input are similar and that their differences are within a 4-percent range. The staff reviewed the applicant's response and finds it to be acceptable because it demonstrated that FSI effects for the case of the CSDRS-HF and soil type 9 are enveloped by those for the CSDRS and soil types 7, 8, and 11. In the proposed markups included in the RAI response, the applicant incorporated the summary of relevant portions of the RAI response into DCA Part 2, Tier 2, Section 3.7.2.

- The applicant indicated that the SASSI2010 model can incorporate only the lumped fluid masses and does not have an FSI analysis capability. Consequently, the applicant used the ANSYS model to compute the hydrodynamic pressures while accounting for the full FSI effects. Then, it added the difference from the SASSI2010 and ANSYS models to the SAP2000 RXB model as equivalent static loads to account for the missing FSI effects. The applicant determined that it needed to apply an average pressure of 4.2 psi to the pool walls and foundation to account for the missing FSI hydrodynamic effects. However, instead of directly applying 4.2 psi to the pool walls and foundation, the applicant took an 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 4.2-psi average hydrostatic pressure. In addition, the applicant proposed to include a new COL information item (COL Item 3.7-12) that 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 response and accompanying DCA markups and finds them 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 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 3–7, 2018 (ADAMS Accession No. ML19098A162), 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. Based on this review, the staff is tracking **RAI 8934**, **Question 03.07.02-15**, as **Confirmatory Item 03.07.02-2,** pending the applicant's update to the next revision of the DCA.

ANSYS Fluid-Structure Interaction Model

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.2.4, describes an ANSYS model used to calculate the hydrodynamic pressures on the reactor pool walls and foundation while accounting for the FSI effects. The staff identified issues that the applicant needed to clarify concerning the RXB ANSYS model and issued **RAI 8934**, **Question 03.07.02-13**, with the following specific requests:

- The applicant indicated that the COMBIN14 spring elements represent the bottom nodes of the foundation but did not provide details about these spring elements. Therefore, the staff requested the applicant to describe how it determined and evaluated the spring constant for COMBIN14 elements.

- The applicant stated that the ANSYS model uses fixed-base boundary conditions; however, the same DCA section describes the foundation bottom nodes as spring elements. Therefore, the staff requested the applicant to explain how fixed-base boundary conditions are realized by using spring elements.

- DCA Part 2, Tier 2, Figure 3.7.2-34, shows a plan view of the wall segments used for the FSI analysis. The figure also shows the walls separating the refueling pool, spent fuel pool, and dry dock. However, the ANSYS finite elements presented in DCA Part 2, Tier 2, Figure 3.7.2-33, do not show a wall segment separating the refueling pool and spent fuel pool and a segment separating the refueling pool and dry dock. Therefore, the staff requested the applicant to clarify this discrepancy.

- The applicant stated that the input to the ANSYS analysis is the CSDRS-compatible Capitola time history. In Section 8.0 of TR-0916-51502-P, "NuScale Power Module

Seismic Analysis," (ADAMS Accession No. ML18271A181) dated September 28, 2018 the applicant indicated that it obtained the time history input to the NPM seismic analysis from the RXB SASSI2010 analysis based on the CSDRS-compatible Capitola time history and soil type 7. However, the applicant did not explain how it selected the CSDRS-compatible Capitola time history and soil type 7 and why they are adequate. Therefore, the staff requested the applicant to justify the adequacy of the NPM seismic analysis based on a single CSDRS-based time-history and a single soil type, whereas the seismic design of the RXB is based on an analysis involving multiple time histories and soil types, as discussed in DCA Part 2, Tier 2, Section 3.7.2.4.

In its response dated October 3, 2017 (ADAMS Accession No. ML17277A300), to **RAI 8934**, **Question 03.07.02-13**, the applicant provided the following information:

- The applicant stated that the spring elements at the base of the ANSYS model have very high stiffness values to represent a fixed-base condition and, therefore, the spring constant for COMBIN14 elements do not need to be evaluated. The bottom of the foundation basemat of the RXB ANSYS model has three COMBIN14 spring elements attached to each node with a stiffness value of 1×10^8 pound-force per inch for the E-W, N-S, and vertical directions. The applicant also proposed markups for DCA Part 2, Tier 2, Section 3.7.2.1.2.4, to provide additional details of these spring elements used in the ANSYS model. The staff finds that the applicant's response and accompanying markups adequately addressed its question on spring elements and is acceptable.

- The applicant stated that it used fixed-base boundary conditions by connecting the nodes at the bottom of the base to boundary condition nodes using three orthogonal 0.1-inch-long COMBIN14 spring elements in the X, Y, and Z directions. These boundary condition nodes are fixed in translation in the direction of the attached spring element and free in all other degrees of freedom. Because the COMBIN14 spring elements have very high stiffness values, the assembly of the COMBIN14 spring element to a fixed boundary becomes a fixed boundary. The applicant also proposed markups for DCA Part 2, Tier 2, Section 3.7.2.1.2.4, to provide additional details of how the fixed-base boundary conditions are realized by using spring elements. The staff finds that the applicant's response and accompanying markups adequately addressed its question on boundary conditions involving the spring elements and is acceptable.

- The applicant stated that the ANSYS model includes a wall segment separating the refueling pool and spent fuel pool and a segment separating the refueling pool and dry dock; however, they are hidden by the fluid elements shown in DCA Part 2, Tier 2, Figure 3.7.2-33. The applicant also provided a new figure, DCA Part 2, Tier 2, Figure 3.7.2-34, that it will add to DCA Part 2, Tier 2, Section 3.7.2, to show these details. The staff finds that the applicant's response adequately addressed its question and provided DCA markups that are acceptable.

- The applicant stated that the analysis of the NPM based on the CSDRS-compatible Capitola time history and soil type 7 is adequate to demonstrate that the NPM design is acceptable and meets the requirements of GDC 2 and 10 CFR Part 50, Appendix S, at sites with characteristics consistent with these inputs. The applicant further stated that NuScale is seeking certification of the NPM design for this single time history and soil type, whereas the RXB analysis described in DCA Part 2, Tier 2, Section 3.7.2, encompasses additional soil profiles. The staff finds the applicant's response to be acceptable because the applicant specifically stated that it seeks certification of the NPM design for the applicant-chosen single time history and soil type. The staff also confirmed that the applicant incorporated its proposed markups into DCA Part 2, Tier 2, Section 3.7.2.

Based on its review of the applicant's response, **RAI 8934**, **Question 03.07.02-13**, is resolved and closed.

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.2.4, describes the FSI analysis for the reactor pool using the ANSYS model and discusses the results. However, the staff identified issues that the applicant needed to address and issued **RAI 8934**, **Question 03.07.02-14**, with the following specific requests:

- The applicant described the modeling of the contact between the bottom nodes of the CNV and pool foundation surface. However, the applicant did not address the frictional nature of the contact. Therefore, the staff requested the applicant to clarify whether friction is considered between the contacting surfaces and to provide the coefficient of friction if it is considered.

- DCA Part 2, Tier 2, Figure 3.7.2-32, shows that the ANSYS model includes backfill soil elements. However, the DCA does not provide the boundary conditions of the model involving backfill. Therefore, the staff requested the applicant to describe the boundary conditions on the exterior sides of the backfill soil elements and to indicate whether they are modeled as a stress-free surface.

- DCA Part 2, Tier 2, Figure 3.7.2-35, indicates perfect matching between accelerations for wall segments X1 and X3, whereas DCA Part 2, Tier 2, Figure 3.7.2--37, indicates a slight difference between pressures for wall segments X1 and X3. The staff requested the applicant to explain the difference in pressure between wall segments X1 and X3, although their accelerations are shown to match perfectly.

- DCA Part 2, Tier 2, Figures 3.7.2-37 and 3.7.2-38, indicate a nonzero pressure at the top of the curve, which appears to indicate that the top endpoint of the curve does not correspond to the free water surface of the pool top. Therefore, the staff requested the applicant to clarify and explain why the curves do not cover the elevations all the way up to the free water surface.

In its response dated October 3, 2017 (ADAMS Accession No. ML17277A300), to **RAI 8934**, **Question 03.07.02-14**, the applicant provided the following information:

- The applicant stated that DCA Part 2, Tier 2, Section 3.7.2.1.2.4, describes the modeling of contact between the bottom nodes of the CNV and pool foundation surface and that the model does not consider the friction between the bottom nodes of the CNV and pool foundation surface.

- The applicant stated that the boundary conditions on the exterior sides of backfill soil elements are set up such that the backfill soil is free around the perimeter (i.e., a stress-free surface at four vertical sides) and fixed at the bottom. Each node at the bottom surface of the backfill soil is attached to three orthogonal COMBIN14 elements with a length and spring stiffness of 0.1 inch and 1×10^8 pounds per inch (lb/in.), respectively.

- The applicant stated that the difference in pressure between wall segments X1 and X3 in DCA Part 2, Tier 2, Figure 3.7.2-35, is the result of the asymmetric geometry of the pool and that the hydrodynamic pressure (calculated from the 3-D FSI analysis) is dependent on both the seismic input (i.e., acceleration) and geometry of the pool.

- The applicant stated that the hydrodynamic pressure plots in DCA Part 2, Tier 2, Figures 3.7.2-37 and 3.7.2-38, are based on the pressure values at the centroid of each water element of the model, and these values were then assigned at the top coordinate of the element height (i.e., the centroidal pressure values are used over the full element height). This assigns a pressure to the free surface of the pool and results in the horizontal shift as seen in the figures.

The staff evaluated the applicant's response and finds it to be acceptable because it provided an adequate explanation and information that clarified the staff's questions about details of the RXB ANSYS model and about the analysis results from the model. Based on this review, **RAI 8934**, **Question 03.07.02-14**, is resolved and closed.

Verification and Validation of Computer Programs

DCA Part 2, Tier 2, Revision 0, Section 3.7.5, describes the computer programs used in the analysis of the site-independent NuScale seismic Category I and Category II structures. However, it does not provide information on the V&V of these programs needed for the staff to determine the adequacy of the programs for use in the NuScale application. Therefore, in **RAI 8936**, **Question 03.07.02-7**, the staff requested the applicant to provide information summarizing the V&V of the computer programs used to calculate design-basis seismic demands for the site-independent structures of the NuScale standard design. The demonstration should test those characteristics of the software that mimic the physical conditions and material properties that represent the NuScale design in the numerical analysis. In addition, the V&V should cover the full range of parameters used in the NuScale design-basis seismic demand calculations. The staff recognizes that the ANSYS and SAP2000 computer programs are generally accepted commercial finite element codes that have been widely evaluated with respect to their capabilities and performance by the engineering community, including nuclear applications; therefore, the staff's review focused on the V&V of the SASSI2010 computer program.

In its response dated January 31, 2018 (ADAMS Accession No. ML18031B204), and its supplemental response dated January 11, 2019 (ADAMS Accession No. ML19011A347), to **RAI 8936**, **Question 03.07.02-7**, the applicant provided information on the V&V of the SASSI2010 model that was used for seismic demand calculations for NuScale standard plant structures subjected to the SSE vibratory ground motions. 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. In its response to **RAI 8936**, **Question 03.07.02-7**, 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 hertz 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 applicant also provided DCA markups that capture key elements of SASSI2010 V&V.

The staff reviewed the applicant's RAI response and accompanying markups and finds them 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 generally accepted industry practice. The staff also confirmed that the applicant incorporated the proposed markups into DCA Part 2, Tier 2, Section 3.7.5. Based on this review, **RAI 8936**, **Question 03.07.02-7**, is resolved and closed.

Reactor Building and NuScale Power Module Support Interface Loads

DCA Part 2, Tier 2, Revision 0, Appendix 3A, Section 3A.1, references TR-0916-51502-P for NPM seismic analysis and indicates that a detailed dynamic analysis of the NPM subsystem was performed using a detailed 3-D NPM model with acceleration time histories from the SASSI RXB model as the input, in order to capture the coupling effects between the RXB and NPMs. The staff sought to confirm that the seismic demands at the NPM supports from the SASSI RXB model and from the ANSYS 3-D NPM model reasonably match; however, the DCA and TR-0916-51502-P do not provide such information. Therefore, in **RAI 8936**, **Question 03.07.02-10**, the staff requested the applicant to compare the seismic load demands at the NPM upper and bottom support locations interfacing with the RXB obtained from the SASSI RXB model and the ANSYS 3-D NPM model. The staff also requested the applicant to explain any significant differences and confirm that the loads used for the NPM support designs are conservative.

In its response dated June 25, 2018 (ADAMS Accession No. ML18176A155), to **RAI 8936**, **Question 03.07.02-10**, and in its supplemental responses dated September 6, 2018 (ADAMS Accession No. ML18249A413), and October 22, 2018 (ADAMS Accession No. ML18295A778), the applicant provided information comparing the reaction forces at the NPM skirt and lug restraints from the SASSI analysis of the RXB model and from the ANSYS analysis of the detailed 3-D NPM model and indicated that the results from the ANSYS analysis exceeded those from the SASSI analysis at some locations. At the interface between the NPMs and the RXB, the design loads for the skirt supports are defined as the envelope of the SASSI RXB model and the 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 SASSI building model and ANSYS 3-D model, as described in more detail in DCA Part 2, Tier 2, Appendix 3B, Section 3B2.7. In updated DCA Part 2, Tier 2, Table 3B-28, the applicant provided the enveloping skirt and lug reactions obtained from both the NPM seismic analysis and the SASSI RXB seismic analysis using soil type 7 and the CSDRS-compatible Capitola input motion.

In its response to **RAI 8911, Question 03.09.02-43** dated January 31, 2019 (ADAMS Accession No. ML19031C983), the applicant described that it performed a new set of seismic runs involving the RXB-NPM interfaces taking 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). The new runs also incorporated an enhanced methodology for modeling hydrodynamic mass in the pool area. However, the applicant provided information on the maximum seismic forces computed at NPM support locations that are lower than the previously provided values in its response to **RAI 8936**, **Question 03.07.02-10**. Therefore, the staff requested the applicant to provide in its follow-up response an explanation for such changes to the forces at the NPM supports. The applicant was also requested to provide information on the hydrodynamic pressure loads on the pool walls determined from new analyses in comparison with the previously provided values. Further, the staff requested the applicant to provide updated DCA markups in Sections 3.7.2 and 3.8.4 for those portions affected by the information provided in the response to **RAI 8911, Question 03.09.02-43**. The information on RXB-NPM interface support loads provided in the response to **RAI 8936, Question 03.07.02-10** will be updated by new information provided in a follow-up response to **RAI 8911, Question 03.09.02-43.** Therefore, **RAI 8936, Question 03.07.02-10**, is **Open Item 03.07.02-1**.

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, including the natural frequencies and modal mass ratios. Because SASSI2000 uses a complex frequency response analysis method, the applicant used the corresponding SAP2010 models with a fixed-base boundary condition to generate the modal properties. The staff notes that the RXB and CRB FEMs were developed using SAP2000 models, which serve as the master models. The FEMs used with SASSI2010 and ANSYS were created from the SAP2000 models.

<u>Seismic Demand at Critical Sections</u>

In DCA Part 2, Tier 2, Revision 0, Tables 3.7.2-23, 3.7.2-24, and 3.7.2-25, the applicant provided an SSI analysis results for one particular shell, beam, and solid element, respectively. However, the staff notes that the applicant did not provide analysis results at other key locations. Therefore, in **RAI 8935**, **Question 03.07.02-25**, the staff requested the applicant to provide the design-basis seismic demands (e.g., forces, moments, soil pressures, accelerations, displacements, ISRS), at all applicable critical section locations of the RXB and CRB, used in the structural design evaluations in DCA Part 2, Tier 2, Sections 3.8.4 and 3.8.5.

In its response dated December 20, 2018 (ADAMS Accession No. ML18354B049), to **RAI 8935**, **Question 03.07.02-25**, the applicant provided the requested information, including a set of tables and figures of seismically induced accelerations, ISRS, soil pressures, displacements, and forces and moments, at critical section locations used in the structural design evaluations of the RXB and CRB in DCA Part 2, Tier 2, Sections 3.7.1, 3.7.2, 3.7.3, 3.8.4, and 3.8.5. The staff reviewed the applicant's response and finds it to be acceptable because it provides the requested information and its locations in the DCA concerning the design-basis seismic demands at all applicable locations used in the structural design evaluation of the RXB and CRB. Based on this review, the staff considers **RAI 8935**, **Question 03.07.02-25**, as resolved and closed.

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. 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.

<u>Mesh Discretization</u>

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 recommends that the element mesh size should be selected on the basis that a further refinement has only a negligible effect on the solution results. However, Revision 0 of the DCA did not discuss this issue. Therefore, in **RAI 8932**, **Question 03.07.02-1**, the staff requested the applicant to provide information that demonstrates the adequacy of the finite elements used in the RXB and CRB standalone models and the triple building model.

In its response dated December 21, 2017 (ADAMS Accession No. ML17355A677), and in its supplemental response dated April 3, 2018 (ADAMS Accession No. ML18093B061), to **RAI 8932**, **Question 03.07.02-1**, 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. Therefore, there is no need to study the effects of the mesh refinement on the SSI, ISRS, or SSSI. 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 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 applicant also provided markups that summarize the information in its supplemental RAI response.

The staff reviewed the applicant’s response 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. Further, during the regulatory audit from December 3–7, 2018 (ADAMS Accession No. ML19098A162), the staff reviewed the detailed calculations and conclusions from the mesh sensitivity study and confirmed that they are consistent with the information in the RAI response and in the markups of DCA Part 2, Tier 2, Section 3.7.2. The staff also confirmed that the applicant incorporated its proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Based on this review, **RAI 8932**, **Question 03.07.02-1**, is resolved and closed.

<u>Presence of Coarse Finite Elements</u>

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.1.3, describes the SASSI2010 computer code and states the following:

The building models have element sizes that are similar to the 6.25 feet layers that were used to determine the wave passage frequency of the soil. There are instances where development of the model required individual elements to have a dimension as large as 12 feet in the RXB and as large as 20 feet in the CRB. However, the typical element size is approximately 6 feet. Therefore, the wave passage frequencies of both buildings is above the cutoff frequencies used for the analysis.

However, the staff notes that the applicant did not provide information on the potential effects of these coarse finite elements on the calculated seismic demands of the structures. Therefore, in **RAI 8932**, **Question 03.07.02-2**, the staff requested the applicant to provide more details about the elements that have a dimension of 12 feet or 20 feet, including their locations in the building, and to demonstrate that the presence of these coarse elements will not affect the results of seismic demand analyses for the RXB and CRB.

In its response dated September 27, 2017 (ADAMS Accession No. ML17271A239), and in its supplemental response dated December 6, 2017 (ADAMS Accession No. ML17340B393), to **RAI 8932, Question 03.07-02-2**, the applicant stated that, in the CRB model, the elements with large dimensions (up to 20 feet) are nonstructural membrane elements used for the purpose of applying wind loads to the steel beams and columns of the steel-framed structure above elevation 120′ and that these coarse elements are not present in the seismic analyses and therefore will not affect the seismic demand results. The applicant also stated that the RXB model has 24 solid elements with a maximum dimension of 12 feet at the pool floor, and this mesh transitions into the uniform soil mesh at the bottom of the basemat with an average element size of 6.25 feet. The applicant further stated that the single layer of coarse basemat transition elements have minimal effect on the seismic analysis results.

The staff reviewed the applicant’s response and finds it to be acceptable because (1) the membrane elements with a maximum dimension of 20 feet 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 12 feet 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 6.25 feet. The staff also confirmed that the applicant incorporated its proposed markups that capture relevant information in the RAI response into DCA Part 2, Tier 2, Section 3.7.2. Based on this review, **RAI 8932**, **Question 03.07.02-2**, is resolved and closed.

<u>Passing and Cutoff Frequency</u>

DC/COL-ISG-01 states that structural models used for SSI calculations should be modeled in sufficient detail such that they contain frequencies of up to at least 50 hertz. This is intended to address the fact that many sites in the eastern United States are characterized by high-frequency earthquake ground motions; therefore, analytical models should be capable of capturing responses that would occur at these high-frequency sites. DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.1.3, discusses cutoff frequencies associated with SSI analyses performed using the SASSI2010 models. However, the staff identified issues that the applicant needed to address and issued **RAI 8932**, **Question 03.07.02-3**, with the following requests:

- In DCA Part 2, Tier 2, Section 3.7.2.1.1.3, the applicant stated that “For the analysis of Soil Types 7, 8 and 11 with the CSDRS the cut-off frequency was established at 52 Hz. This is higher than the wave passing frequency of the soft soil profile (Soil Type 11) but less than the passing frequency of the other two soils (see Table 3.7.1-20)." However, DCA Part 2, Tier 2, Table 3.7.1-20, shows 12 hertz as the passing frequency for soil type 11, and DCA Part 2, Tier 2, Tables 3.7.2-18 and 3.7.2-19, indicate that TFs are calculated for frequencies up to 52 hertz (cutoff frequency) for soil type 11. Therefore, the staff requested the applicant to justify the validity of the TF calculations for frequencies beyond the passing frequency for soil type 11.

- In DCA Part 2, Tier 2, Section 3.7.2.1.1.3, the applicant stated that "For the analysis of Soil Types 7, 8 and 11 with the CSDRS, the cut-off frequency was established at 52 Hz." However, the staff finds that DCA Part 2, Tier 2, Table 3.7.2-19, used a cutoff frequency of 72 hertz for Soil Type 7 with the CSDRS. Therefore, the staff requested the applicant to clarify this inconsistency. In addition, in DCA Part 2, Tier 2, Section 3.7.2.1.1.3, the applicant further stated that "For the analysis with the rock profiles (Soil Type 7 and 9) and the CSDRS-HF, the cut-off frequency was established at 72 Hz." However, DCA Part 2, Tier 2, Table 3.7.2-21, used a cutoff frequency of 52 hertz for soil type 9 with the CSDRS-HF. Therefore, the staff requested the applicant to clarify this inconsistency.

In its response dated September 27, 2017 (ADAMS Accession No. ML17271A239), and in its supplemental response dated December 6, 2017 (ADAMS Accession No. ML17340B393), to **RAI 8932**, **Question 03.07.02-3**, the applicant provided the following information to address the questions that the staff requested about the SASSI2010 analysis methods:

- In response to the staff's request to justify the validity of TF calculations for frequencies beyond the passing frequency for soil type 11 (soft soil), the applicant indicated that the SSI effects will be filtered out at frequencies higher than the passing frequency and may result in a damped structural response for soil type 11. The applicant further indicated that this will not negatively affect the overall TF calculations in the high-frequency range because they will be covered by firm and hard rock soil cases. The staff finds the applicant's explanation that the calculation of TFs in the high-frequency range beyond the passing frequency (12 hertz) of soil type 11 is not a concern because calculations in the high-frequency range are effectively covered by soil profiles that are stiffer than soil type 11 to be acceptable.

- In response to the staff's request to clarify the discrepancies between the cutoff frequencies for TF calculations, as stated in DCA Part 2, Tier 2, Section 3.7.2.1.1.3, and as indicated in DCA Part 2, Tier 2, Tables 3.7.2-18 through 3.7.2-21, the applicant provided some technical details to explain such discrepancies. Also, inDCA Part 2, Tier 2, Section 3.7.2.1.1.3, the applicant proposed a revised statement that the cutoff frequency was established at 52 hertz for analyses for soil types 7, 8, and 11 with the CSDRS and at 72 hertz for analyses for soil types 7 and 9 with the CSDRS-HF. The applicant also proposed to add notes to DCA Part 2, Tier 2, Tables 3.7.2-19 and 3.7.2-21 to further clarify cutoff frequencies associated with different soil types and ground motions. The staff finds the applicant's explanation and proposed DCA markups to be acceptable because they address identified discrepancies associated with cutoff frequencies.

The staff confirmed that the applicant incorporated the proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Based on this review, **RAI 8932**, **Question 03.07.02-3**, is resolved and closed.

Reactor Building Foundation Modeling, Rigid Spring Constants, and Aspect Ratio of Finite Elements

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.1.3, discusses other aspects of the RXB SASSI2010 model. However, the staff identified issues that the applicant needed to address and issued **RAI 8933, Question 03.07.02-17,** with the following requests:

- DCA Part 2, Tier 2, Figure 3.7.2-13, indicates that the refueling area foundation is lower than the neighboring reactor pool and spent fuel pool area. DCA Part 2, Tier 2, Section 3.7.2.1.2.1, indicates that the refueling area foundation is 6 feet lower than the neighboring reactor pool foundation. However, the staff finds that DCA Part 2, Tier 2, Figure 3.7.2-20, does not indicate these differences in the foundation elevation. Therefore, the staff requested the applicant to clarify whether the RXB model accounted for these foundation elevation differences and, if it does not, to justify why it did not do so.

- In DCA Part 2, Tier 2, Section 3.7.2.1.2.1, the applicant stated that “The rigid springs have a zero length and have a stiffness value large enough to simulate rigid connection. The large stiffness used is arbitrarily chosen to be ten billion lbs per inch, or 10^{10} lbs/inch, in the three global directions.” To model the spring as a rigid spring, the value of its spring constant should be sufficiently larger than the stiffness of the structural element (basemat) to which it is attached. Therefore, the staff requested the applicant to confirm the adequacy of the number (1×10^{10} lb/in.) chosen for the spring constant by comparing it to the stiffness of the adjacent basemat element or by performing an appropriate sensitivity analysis using a number at least an order of magnitude different.

- DCA Part 2, Tier 2, Table 3.7.2-1, shows the maximum aspect ratio for RXB finite elements to be 11.9. The staff requested the applicant to ensure that this value of aspect ratio is within the range of the parameters covered in the SASSI V&V and, if it is not, to justify the adequacy of using the maximum aspect ratio of 11.9 for RXB finite elements.

In its response dated December 3, 2018 (ADAMS Accession No. ML18337A441), to **RAI 8933**, **Question 03.07.02-17**, the applicant provided the following information:

- DCA Part 2, Tier 2, Figure 3.7.2-13, displays the south side of the RXB, whereas DCA Part 2, Tier 2, Figure 3.7.2-20, displays the north side of the RXB and a section cut that does not include the refuel or spent fuel pool. The pits are designed as an integral part of the entire foundation design, and the pits are necessary for plant operational and maintenance needs during refueling outages. The applicant also stated that the RXB model did not account for elevation differences in the refueling pool because the sides of the pit transition into the main foundation with sloped sides, thus minimizing the resistance during horizontal motion, and because the base areas of the pits are small compared to the overall foundation size. Further, the bottom thicknesses of the pits remain the same as the foundation thickness (i.e., 10 feet), which is greater than the elevation difference of 6 feet. The staff finds the applicant’s response to be acceptable because the base areas of the pits are small compared to the overall foundation footprint; therefore, the difference in foundation elevation in a limited number of structural elements would not change the overall structural dynamics or response of the RXB.

- The applicant performed a sensitivity analysis by increasing the stiffness of the rigid springs by an order of magnitude (i.e., $1x10^{11}$ lb/in.) and by comparing the results with those obtained from the base case (i.e., with the rigid spring stiffness equaling $1x10^{10}$ 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 finds the applicant's response to be acceptable, and the value of $1x10^{10}$ lb/in. used 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.

- Most elements in the finite element mesh are square and, therefore, have an aspect ratio close to 1.0. The aspect ratio of 11.9 is for nonstructural, surface elements. The staff finds the applicant's response to be acceptable because the exceptional high aspect ratio applies to a limited number of nonstructural elements that would not affect the response of the structure.

The staff also reviewed the proposed markups of DCA Part 2, Tier 2, Section 3.7.2.1.2.1 and Table 3.7.2-1 and finds them to be acceptable because they capture the findings from the sensitivity study on the rigid springs and provides a note on the aspect ratio as evaluated above. Based on this review, the staff is tracking **RAI 8933**, **Question 03.07.02-17**, as **Confirmatory Item 03.07.02-3,** pending the applicant's update to the next revision of the DCA.

<u>Location of the Reactor Building Crane When Not in Use</u>

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.2.3, states that "When not in use, the Reactor Building Crane (RBC) is parked over the refueling pool with the trolley at the north end near the dry dock gate." However, that same DCA section also states that "For the analysis of the RXB, the RBC is unloaded (i.e., no suspended NPM) and located in the middle of the reactor pool area as shown in Figure 3.7.2-24." In **RAI 8933**, **Question 03.07.02-18**, the staff requested the applicant to explain why, in developing the seismic analysis model for the RXB, it located the unloaded RBC in the middle of the reactor pool area instead of locating it over the refueling pool where it is parked when not in use.

In its response dated October 3, 2017 (ADAMS Accession No. ML17277A312), to **RAI 8933**, **Question 3.07.02-18**, the applicant stated that the seismic analysis model for the RXB considered the locations for the wheels of the RBC to be approximately in the middle region of the N-S pool walls, and that this covers the most flexible region of the pool walls to maximize structural effects during a seismic event. The staff finds the applicant's response acceptable because positioning the RBC in the middle of the reactor pool area (the most flexible region) produces maximum structural loading effects during a seismic event. Based on this review, **RAI 8933**, **Question 03.07.02-18**, is resolved and closed.

<u>Comparison of the SAP2000 and SASSI 2010 Models</u>

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.2.6, states that "In the calculation of the structural frequencies for comparison, the structure is assumed to be surface founded in both the SAP2000 and SASSI2010 analyses. In the SASSI2010 analysis, the backfill soil was also assumed to be seated on top of a rigid halfspace with the structure." The same DCA section further states, "However, the effect of backfill soil is more accurately captured in the SASSI2010 transfer functions than in the modal analysis of SAP2000." DCA Part 2, Tier 2, Tables 3.7.2-10 and 3.7.2-11, provide the results of the SAP2000 and SASSI2010 model comparisons for the RXB and CRB, respectively. However, DCA Part 2, Tier 2, Table 3.7.2-11, presents the comparisons only for higher modes (i.e., N>72). Therefore, in **RAI 8933**, **Question 03.07.02-19**, the staff requested the applicant to explain (1) how the SAP2000 and SASSI2010 models incorporated the backfill soils and (2) why DCA Part 2, Tier 2, Table 3.7.2-11, does not include the comparison of structural frequencies for lower modes.

In its response dated October 3, 2017 (ADAMS Accession No. ML17277A312), and in its supplemental response dated December 6, 2017 (ADAMS Accession No. ML17340B394), to **RAI 8933**, **Question 03.07.02-19**, the applicant provided the following information:

- 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 that the backfill soil is free around the perimeter and fixed at the bottom. The applicant also stated that, in order to verify the modeling approach, total weights and fixed-base modal frequencies obtained from SAP2000 and SASSI2010 models are computed and compared. For the SAP2000 model, the modal frequencies are obtained by performing modal analysis; for the SASSI2010 model, because SASSI2010 does not perform modal analysis, 1g transient analyses in the X, Y, and Z directions are performed and peaks of the TFs are computed representing dominant structural frequencies. The applicant further stated that the SASSI2010 modeling results are more accurate because the SASSI2010 TFs include the effects of structural damping, while the SAP2000 modal frequencies are independent of the structural damping.

- The applicant stated that the CRB fixed-base model frequency comparisons are made at critical locations where maximum displacements are expected to occur and that DCA Part 2, Tier 2, Table 3.7.2-11, lists these critical locations. Because SASSI2010 does not perform modal analysis, frequencies corresponding to peaks of TFs at the critical locations are compared with the SAP2000 modal analysis. The applicant illustrated that, at roof center, the SASSI output is compared with the 72^{nd} mode, whose modal frequency matches the frequency at the peak of the TF.

The staff reviewed the applicant's response and finds it to be acceptable because it explains the difference in modeling approaches between the SAP2000 and SASSI2010 computer programs and provided proposed DCA markups that summarize the information in its RAI response. The staff confirmed that the applicant incorporated the proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Based on this review, **RAI 8933**, **Question 03.07.02-19**, is resolved and closed.

<u>Validation of the Simplified NuScale Power Module Beam Model</u>

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.2.2, describes the NPM model included in the RXB SAP2000 and SASSI2010 models. The applicant indicated that the NPM beam model used in the SAP2000 and SASSI2010 building models is derived from the corresponding NPM detailed 3-D model developed in ANSYS. However, the staff had a concern about whether the NPM beam model is adequately developed and dynamically compatible with the corresponding detailed 3-D model. Therefore, in **RAI 9114**, **Question 03.07.02-31**, the staff requested the applicant to explain how it developed the NPM beam model, which is included in the SAP2000/SASSI2010 RXB models, and how it validated NPM beam model's dynamic compatibility with the detailed 3-D NPM model in the ANSYS computer code.

In its response dated November 13, 2017 (ADAMS Accession No. ML17317B553), and in its supplemental responses dated February 21, 2018, and October 31, 2018 (ADAMS Accession No. ML18304A261), to **RAI 9114**, **Question 03.07.02-31**, the applicant indicated that the simplified beam model was developed to have similar dynamic characteristics as those of the 3-D ANSYS model documented in TR-0916-51502-P. 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. The applicant also provided markups indicating that TR-0916-51502-P will include a table comparing the dynamic properties of the NPM beam model and the 3-D model and that DCA Part 2, Tier 2, Section 3.7.2, will include an assessment of how the comparison demonstrates dynamic compatibility between the two models. The staff reviewed the applicant's RAI response and the accompanying DCA markups and finds them acceptable because the NPM beam model included in the SAP2000 and SASSI2010 RXB models 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 (ADAMS Accession No. ML19098A162), 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 applicant's response to **RAI 9114**, **Question 03.07.02-31**, and in the DCA markups. Based on this review, the staff is tracking **RAI 9114**, **Question 03.07.02-31**, as **Confirmatory Item 03.07.02-4,** pending the applicant's update to the next revision of the DCA.

Boundary Conditions of the NuScale Power Module Supports

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.2.2, discusses the boundary conditions for the NPM supports, including the lug restraints and skirt. However, the staff identified technical issues that need to be addressed by the applicant and issued **RAI 8933**, **Question 03.07.02-20**, with the following requests:

- The note at the bottom of DCA Part 2, Tier 2, Figure 3.7.2-28, states that the NPM is restrained from horizontal and vertical movements. However, DCA Part 2, Tier 2, Section 3.7.2.1.2.2, indicates that the NPM can move freely in the upward direction. Therefore, the staff requested the applicant to clarify this apparent discrepancy. In addition, the staff requested the applicant to clarify whether the vertical liftoff (under tension) of the bottom of an NPM from the pool floor is allowed and, if vertical liftoff is allowed, to address the effect of the potential impact load of the NPM on the liner plate of the pool floor as a result of vertical seismic motion.

- In addition, DCA Part 2, Tier 2, Section 3.7.2.1.2.2, further indicates that the twist about the vertical axis is released at the base of the NPM model. Therefore, the staff requested the applicant to describe the interface boundary conditions (preferably in a tabular form) at the NPM top and bottom supports in detail and to explain how they are implemented in the ANSYS and SASSI2010 models.

In its response dated October 3, 2017 (ADAMS Accession No. ML17277A312), and in its supplemental response dated December 6, 2017 (ADAMS Accession No. ML17340B394), to **RAI 8933**, **Question 03.07.02-20**, the applicant provided the following information:

- The applicant stated that vertical liftoff from the bottom of an NPM from the pool floor is allowed and that the CNV skirt support is free to move vertically upward and is restrained downward by contact with the pool floor. The applicant also stated that the RXB structural model was created by the computer program SAP2000, which was converted to an identical SASSI2010 RXB model to perform the SSI analyses; however, because SASSI2010 cannot model such nonlinear interface conditions, spring supports were added to stabilize the analysis model. In its markups to DCA Part 2, Tier 2, Figure 3.7.2-28, the applicant added a note stating that actual CNV skirt support is free to move vertically upward and is restrained downward and that, because SASSI2010 cannot model this nonlinear behavior, spring supports are added only to stabilize the analysis model. In its response to the staff's question about the vertical liftoff of the NPM during an earthquake and the potential impact load on the pool floor, the applicant stated that it performed a nonlinear time history analysis of a 3-D ANSYS FEM that found that the maximum uplift was less than 0.25 inches and indicated that the potential impact load would be minimal.

- In its RAI response, the applicant included markups that added two new tables (DCA Part 2, Tier 2, Tables 3.7.2-36 and 3.7.2-37) that outline interface boundary conditions between the NPM beam model and the RXB model for the SASSI2010 and ANSYS analyses. The applicant also proposed to revise DCA Part 2, Tier 2, Section 3.7.2.1.2.2, to state that, for the SASSI2010 NPM beam model, the NPM skirt support restricts horizontal and vertical movements, and to remove the statement about the twist being permitted. The applicant's markups also describe the NPM floor support design in the revised DCA Part 2, Tier 2, Appendix 3B, Section 3B.2.7.3 and Figures 3B-48 to 3B-50.

The staff reviewed the applicant's response and accompanying DCA markups and finds them to be acceptable because they clarify staff's question about vertical liftoff of the NPM and describe the interface boundary conditions at the NPM top and bottom supports. The staff confirmed that the applicant incorporated the proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Based on this review, **RAI 8933**, **Question 03.07.02-20**, is resolved and closed.

<u>Clarification on Modeling Assumptions</u>

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.2.6, discusses the comparison of the SAP2000 and SASSI2010 models. The applicant stated that it obtained the SASSI2010 model data by converting the data of the SAP2000 models. In order to verify that the SAP2000 model has been converted accurately into the SASSI2010 model, it compared the total weights of the two models and the fixed base modal frequencies of the two models. However, the staff found that some of the modeling aspects for SAP2000 and SASSI2010 were not clear and, therefore, issued **RAI 8933**, **Question 03.07.02-21**, with the following requests:

- The applicant stated that "In the calculation of the structural frequencies for comparison, the structure is assumed to be surface founded in both the SAP2000 and SASSI2010 analyses. In the SASSI2010 analysis, the backfill soil was also assumed to be seated on top of a rigid halfspace with the structure." DCA Part 2, Tier 2, Figures 3.8.4-15 and 3.8.4-21, indicate that the SAP2000 model includes the backfill soil; however, the paragraph quoted above does not mention backfill for the SAP2000 analysis. Therefore, the staff requested the applicant to clarify whether the SAP2000 analysis also includes the backfill.

- The staff noted that the SAP2000 models in DCA Part 2, Tier 2, Figures 3.8.4-15 and 3.8.4-21, have "2-node rigid link elements" attached to the bottom of the basemat. The staff requested the applicant to explain how the rigid links provide fixed-base boundary conditions.

- The staff further noted that, in order to simulate a fixed-base boundary condition, the SASSI analysis model is seated on top of a rigid halfspace. The staff requested the applicant to provide the value of shear wave velocity used to simulate the rigid halfspace.

In its response dated October 3, 2017 (ADAMS Accession No. ML17277A312), to **RAI 8933**, **Question 03.07.02-21**, the applicant provided the following information:

- The applicant stated that the SAP2000 analysis includes the backfill soil and that all finite element analysis models for the RXB and the CRB include 25-foot-wide backfill soil around the exterior perimeter of the building. The applicant also stated that DCA Part 2, Tier 2, Figure 3.8.4-15 and Figure 3.8.4-21, correctly show the backfill in the SAP2000 model referred to in DCA Part 2, Tier 2, Section 3.7.2.1.2.6. The staff confirmed that DCA Part 2, Tier 2, Section 3.7.2.1.2.6, includes a statement indicating that the SAP2000 model does indeed include backfill soil.

- The applicant stated that, in earlier design phases, the fixed-base boundary condition was created at the bottom of the basemat using link elements with very large stiffness in three translational degrees of freedom (Ux, Uy, and Uz). However, in the final design phase, these links were eliminated to reduce the number of nodes in the SAP2000 model. In the final model, 3-D pinned supports (with three translational degrees of freedom, fixed) are used at the bottom of the basemat. The applicable figures in DCA Part 2, Tier 2, Section 3.8.4, replace the description "2-node rigid link element" with the description "3-D pinned support." The applicant provided markups that show the changes to DCA Part 2, Tier 2, Figures 3.8.4-15, 3.8.4-16, 3.8.4-17, and 3.8.4-21. The staff finds that the applicant's response and accompanying DCA markups to be acceptable because the fixed-base boundary condition is adequately implemented using the 3-D pinned support.

- The applicant stated that, for the SASSI2010 analyses, a shear wave velocity of 10,000,000 inches per second (833,333 ft/s) is used to simulate the rigid halfspace, which is considered large enough to model rigid layers and therefore is acceptable.

The staff finds that the applicant's response adequately addressed its concerns as discussed above and therefore is acceptable. The staff confirmed that the applicant incorporated the proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Based on this review, **RAI 8933**, **Question 03.07.02-21**, is resolved and closed.

<u>Modeling of Spring Elements</u>

DCA Part 2, Tier 2, Revision 0, Table 3.7.2-1, summarizes the SASSI2010 RXB model, including the various model properties and parametric values used. However, the staff identified technical issues concerning the procedures used for analytical modeling and issued **RAI 8933**, **Question 03.07.02-22**, with the following requests:

- In DCA Part 2, Tier 2, Table 3.7.2-1, the number of rigid springs connecting RXB and excavated free-field soil is indicated as 4,470. However, it was not clear whether the spring elements are considered as linear springs or 3-D springs. Therefore, the staff requested the applicant to clarify whether "rigid springs" refer to linear spring elements (three per node) or 3-D spring elements (one per node).

- DCA Part 2, Tier 2, Tables 3.7.2-10 and 3.7.2-11, provide the results of the SAP2000 and SASSI2010 model comparisons at selected locations within the RXB and CRB. However, the applicant did not explain how it selected these locations. The staff requested the applicant to describe how it selected these locations for comparisons and to justify the adequacy of these selected locations.

- DCA Part 2, Tier 2, Table 3.7.2-12, summarizes the triple building SASSI model. The table presents the number of "interface springs" between the RWB and RXB and between the RXB and CRB. However, the nature and function of these interface springs are not clear to the staff. Therefore, the staff requested the applicant to provide additional information explaining these interface springs.

In its response dated October 3, 2017 (ADAMS Accession No. ML17277A312), to **RAI 8933**, **Question 03.07.02-22**, the applicant provided the following information:

- The applicant stated that the rigid springs identified in DCA Part 2, Tier 2, Table 3.7.2-1, are modeled as 3-D spring elements with stiffness parameters defined in the three global X, Y, and Z directions.

- The applicant stated that the locations for the modal frequency comparisons in DCA Part 2, Tier 2, Tables 3.7.2-10 and 3.7.2-11, were selected as regions sensitive to major vibrations and that this criterion led to a selection of locations such as flexible roofs, slabs, NPMs, and heavily loaded walls. The comparison of modal frequencies obtained at these sensitive locations showed that SAP2000 and SASSI2010 models of the RXB and CRB are in good agreement.

- The applicant stated that the interface springs in DCA Part 2, Tier 2, Table 3.7.2-12, connect the backfill soil elements interfacing between the buildings and that these springs only exist in the backfill soil elements along the contact surfaces between the RWB and RXB, and between the RXB and CRB. These interfacing springs are not included in the 7P ESM interaction nodes.

The staff finds that the applicant's response to be acceptable because it explained the nature of rigid springs and interface springs used in the SASSI models and because the locations selected for the SAP2000 and SASSI2010 model comparison included key locations sensitive to major vibrations. Based on above review, **RAI 8933**, **Question 03.07.02-22**, is resolved and closed.

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, Revision 0, 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.

<u>Benchmarking Extended Subtraction Method with Direct Method (SASSI)</u>

The 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. In order to discuss the adequacy of the 7P ESM, the applicant conducted a sensitivity study with a 9P ESM, which includes two additional intermediate planes of interaction nodes, and compared the responses from the 7P and 9P ESMs. However, the staff recognized that comparing the 7P ESM with the 9P ESM captures only an incremental enhancement between the two models and that the adequacy of a 7P ESM should be established by benchmarking it against the DM, which produces theoretically accurate results. Therefore, in **RAI 8932, Question 03.07.02-4**, the staff requested the applicant to conduct a benchmark test and provide a comparison of the TFs and seismic responses from the 7P ESM and DM at key locations in the RXB and CRB.

In its response dated April 30, 2018 (ADAMS Accession No. ML18120A261), and in its supplemental responses dated June 6, 2018 (ADAMS Accession No. ML18157A262), and August 24, 2018 (ADAMS Accession No. ML18236A927), to **RAI 8932, Question 03.07.02-4,** the applicant compared acceleration TFs and other seismic responses from the DM and 7P ESM, including ISRS, soil pressures, forces and moments, and relative displacements at key locations of the RXB and CRB. Application of the DM for the SASSI analysis of the full RXB model required the use of a number of interaction nodes (28,830) that exceeded the SASSI2010 program limit of 20,000. Therefore, the applicant used a half model to obtain the results by the DM. The applicant reported that the ISRS calculated by the RXB and CRB 7P ESM are within 15 percent of those calculated by the DM, and exceedances were observed at narrow frequency bands around ISRS peak locations. The applicant also reported that the TF shapes show a good agreement between the 7P ESM and DM, except at a few frequencies where some differences were observed, but they did not significantly affect the analysis results. No spurious peaks were introduced in most of the TFs except for a few spikes, which did not affect ISRS calculations. 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 and soil pressures using the two methods; the comparison showed that the 7P ESM and DM differ as much as 20 percent from each other.

The staff's review of the applicant's RAI response indicated that the results from the DM exceeded the design-basis demands computed using the 7P ESM in certain locations in the RXB and CRB, both in structural forces and moments and in ISRS for equipment qualifications. The applicant demonstrated that the exceedances in structural forces and moments are covered by available design margins in terms of the demand-to-capacity (D/C) ratios for the structural members. However, the RAI response did not adequately address the exceedances in ISRS for equipment qualifications. During the regulatory audit from December 3-7, 2018 (ADAMS Accession No. ML19098A162), the staff noted that the use of the design-basis ISRS without further addressing the exceedances from the sensitivity study may lead to unconservative equipment qualifications. The applicant will address this issue in a followup RAI response and update the DCA as appropriate. The applicant provided an action plan for bringing the issue to closure and the corresponding schedule of completion. Therefore, **RAI 8932, Question 03.07.02-4**, is **Open Item 03.07.02-2**.

Review of Transfer Functions

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.1.1.3, describes the SASSI2010 models used to compute the seismic demands for the RXB and CRB and also discusses TFs generated from these models. The applicant stated that, because of the size and complexity of these models, it is not practical to review TFs at all the nodes in the models. However, the staff determined that a review of the TFs is essential to ensure that the numerical models and their implementation in the SSI analyses are adequate and acceptable. The staff recognizes that the applicant does not have to review TFs at all nodes; however, the applicant should at least evaluate TFs at some key locations. Therefore, in **RAI 8932, Question 03.07.02-5**, the staff requested the applicant to provide information on TFs at key locations in the RXB and CRB and to inspect the TFs to identify whether spurious spikes are present within the frequency range of interest to the SSI analysis. If spikes are present, the applicant should discuss their potential effects on computed seismic demands.

In its response dated November 29, 2018 (ADAMS Accession No. ML18333A387), to **RAI 8932, Question 03.07.02-5**, the applicant presented the plots of acceleration TFs and ISRS at key selected locations in the RXB and CRB from SSI analysis using the cracked concrete, Soil Type 7, and Capitola time history input. The applicant reported that it found spurious spikes in a few TFs, mainly from the built-in interpolation functions in the software. The applicant also indicated that the corresponding seismic input at those frequencies is insignificant; therefore, the corresponding ISRS do not have any spurious peaks. The staff finds the applicant's RAI response and accompanying DCA markups to be acceptable because, based on the ISRS examinations and nonexistence of any spurious peaks in the ISRS, the staff can conclude that the spurious spikes in TFs have no effect on the design of the RXB and CRB or on the ISRS for equipment qualifications. Based on this review, **RAI 8932, Question 03.07.02-5**, is **Confirmatory Item 03.07.02-5**, pending the update to the next revision of the DCA.

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, Revision 0, Section 3.7.2.1.1.3, discusses soil separation and its implementation in the SASSI2010 RXB model. However, the staff identified technical issues the applicant needs to clarify or address and issued **RAI 8932, Question 03.07.02-6**, with the following questions:

a. DCA Part 2, Tier 2, Section 3.7.2.1.1.3, states, "To model the soil separation, the Young's modulus of the backfill elements down to a depth of 25′ (the top four layers of backfill elements) was decreased by a factor of 100." However, the applicant did not provide information as to how the separation depth was determined. Therefore, the staff requested the applicant to provide a basis for 25 feet of separation depth.

b. DCA Part 2, Tier 2, Section 3.7.2.1.1.3, states, "Soil separation has negligible effect on the response of the structure. The primary point of comparison is at the NPM. The study showed that the maximum reaction force at the base of the NPMs decreased by approximately 5 percent, and the maximum reaction force at the NPM lug restraints decreased by more than 15 percent." However, the applicant did not provide information on soil separation effect on transfer functions and seismic demands at other important locations. Review of transfer functions is important because they reflect the behavior of an SSI model and soil separation would induce changes to the transfer functions. Therefore, the staff requested the applicant to provide information on soil separation effect on computed transfer functions and seismic demands at critical locations including the external walls. When soil-separation results increased seismic demands, the applicant was requested to take into account such increased demands in establishing the design basis seismic demands.

c. The staff further notes that a soil-separation study was conducted for the RXB but not for the CRB. Therefore, the staff requested the applicant to provide a technical justification for not conducting a soil-separation study for the CRB.

In its response dated May 22, 2018 (ADAMS Accession No. ML18142C204), and in its supplemental responses dated August 24, 2018, and October 22, 2018 (ADAMS Accession Nos. ML18236A573 and ML18295A791, respectively), to **RAI 8932, Question 03.07.02-6**, the applicant provided information on the sensitivity study performed and its evaluation of the soil-separation effects on seismic demands for the RXB and CRB. The applicant provided the following information to address the staff's questions in the RAI:

a. The applicant chose a soil-separation depth of one-third the embedment depth because the RXB and CRB are deeply embedded structures surrounded by engineered, compacted backfill. In addition, Section 5.1.9 of ASCE 4-16 (2016), "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," indicates a method to address soil separation by assuming no connectivity between the structure and lateral soil over the upper half of the embedment, or 6.01 meters (20 feet), whichever is less. Given the model element sizes, 25 feet was a more convenient depth than the ASCE 4-16 stipulation of 20 feet. The staff considers a soil separation depth of 25 feet to be acceptable because it exceeds the minimum depth of 20 feet specified by a recognized industry standard.

b. The applicant provided spectral acceleration TFs and ISRS at critical locations. Overall, the TFs and ISRS of the two models have very little difference. The response also includes forces at the NPM lug supports, soil pressures on the external walls, maximum shears and moments in exterior walls and two pilasters, and total vertical base reaction results. In some instances, the design loads increase because of soil separation effects. However, the design margins of the RXB structural members are such that these increases do not affect the building design.

c. The applicant conducted a soil-separation study for the CRB and reached conclusions similar to those for the RXB, that is, the spectral acceleration TFs and ISRS at critical locations between the two models is similar, and increases in forces from soil separation are within design margins of the building components, leaving the building design unaltered. The applicant also proposed DCA markups, which include a new COL Item 3.7-11, requiring a COL applicant that references the NuScale Power Plant DC to perform a site-specific analysis that assesses the effects of soil separation and that the COL applicant will confirm that the ISRS in the soil separation cases are bounded by the in-structure response spectra shown in DCA Part 2, Tier 2, Figure 3.7.2-107 through Figure 3.7.2-122.

The staff's review of the applicant's RAI response indicates that the results from the sensitivity study (soil-separated case) exceeded the 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. The applicant demonstrated that the exceedances in structural forces and moments are covered by available design margins in terms of the D/C ratios. The staff's review of the applicant's COL Item 3.7-11 incorporated into the DCA found it to be acceptable because a COL applicant will perform a site-specific analysis that assesses the effects of soil separation on site-specific seismic demands. However, the applicant did not adequately address the exceedances in ISRS for equipment qualifications in its RAI response. During the regulatory audit from December 3-7, 2018 (ADAMS Accession No. ML19098A162), the staff communicated a concern that the use of the design-basis ISRS that does not account for the exceedances from the sensitivity study would constitute an unconservative approach to equipment qualifications. The applicant will address this issue in a followup RAI response and update the DCA as appropriate. The applicant provided an action plan for bringing the issue to closure and the corresponding schedule of completion. Therefore, **RAI 8932, Question 03.07.02-6**, is **Open Item 03.07.02-3**.

<u>Soil-Structure Interaction Analysis for Deeply Embedded Structure</u>

DSRS Section 3.7.2.II.4 provides guidance to consider uncertainties associated with SSI analysis of deeply embedded structures. For instance, for nonvertically propagating shear waves, the applicant can perform a sensitivity evaluation to determine whether this is an important effect to be included in the SSI analysis. However, the staff finds that Revision 0 of the DCA does not provide related information. Therefore, the staff issued **RAI 8935, Question 03.07.02-23**, requesting the applicant to provide information that addresses potential issues pertaining to the seismic SSI analysis of deeply embedded structures of the NuScale standard plant.

In its response dated August 29, 2018 (ADAMS Accession No. ML18241A398), and in its supplemental response dated January 10, 2019 (ADAMS Accession No. ML19010A408), to **RAI 8935, Question 03.07.02-23**, the applicant provided information on the sensitivity study performed for the effects of nonvertically propagating seismic waves on seismic demand calculations. The object 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. For the sensitivity study, the applicant used the RXB model and selected Soil Type 7 for the free-field soil properties because it is a nearly uniform soil profile with a high shear wave velocity of 5,000 feet per second. Using a uniform and stiff soil for this study will give conservative results because, for nonuniform and soft-soil profiles, the angle of incidence decreases as the wave propagates toward the surface because of Snell's law and, thus, the effect of nonvertically propagating waves will be much less. The applicant considered three different angles of incidence: 0 degree (the vertically propagating wave case), 17 degrees, and 30 degrees.

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 a new COL information item that requires a COL applicant referencing the NuScale Power Plant DC to perform a 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. The applicant provided DCA markups that capture a summary of the information in the RAI response.

The staff reviewed the applicant's RAI response and accompanying DCA markups and found them acceptable because (1) combining the coupling responses from nonvertically propagating SV- and P-waves leads to overly conservative and incorrect structural responses, as 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. Based on this review, the staff is tracking **RAI 8935, Question 03.07.02-23**, as **Confirmatory Item 03.07.02-6**, pending the update to the next revision of the DCA.

<u>Seismic Design Basis Soil-Structure Interaction Analysis Cases</u>

According to DCA Part 2, Tier 2, Revision 0, Section 3.7.2.4, there are 540 SSI analysis cases, with five CSDRS-compatible time history inputs, and 72 SSI analysis cases with one CSDRS-HF-compatible time history input, that are considered for seismic demand calculations for the RXB and CRB. However, Section 3.7.2.4 also states that the applicant did not analyze all NuScale seismic Category I SSCs to the same combination of cases. Therefore, the staff issued **RAI 8935, Question 03.07.02-24**, asking the applicant to provide, in a tabular form in the DCA, a summary of the following information with regard to seismic design-basis analyses for the SSCs: (1) analysis cases used to establish the seismic demands (loads and ISRS) for each of the seismic Category I buildings, RXB and CRB, and (2) analysis cases used to establish the seismic demands (loads and ISRS) for the NPM, bioshield, RBC, fuel-handling crane, fuel storage rack, reactor flange tool (RFT), and containment flange tool.

In its response dated October 6, 2017 (ADAMS Accession No. ML17279B156), and in its supplemental response dated December 7, 2017 (ADAMS Accession No. ML17341B655), to **RAI 8935, Question 03.07.02-24**, the applicant stated that the NuScale seismic design basis for all seismic Category I SSC is the CSDRS, as described in DCA Part 2, Tier 2, Section 3.7.1.1.1, and committed in DCA Part 2, Tier 1, Section 3.14.1. The applicant also 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 addition, to clarify seismic analysis cases used to establish seismic demands for the SSCs, the applicant proposed to incorporate two new tables into DCA Part 2, Tier 2 (Tables 3.7.2-33 and 3.7.2-34). The applicant stated that seismic input used for design of the SSCs varies based on different design requirements, locations, and levels of conservatism for each SSC. The seismic design parameters used are classified in Table 3.7.2-33 by eight seismic analysis identification codes, and the applicant revised DCA Part 2, Tier 2, Section 3.7.2.4.6, to include these eight identification codes: (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, and (8) CRB Triple Building Structural Response. DCA Part 2, Tier 2, Table 3.7.2-34, lists the SSCs and notes the accompanying identification codes used for seismic demand analysis. The RAI response also included markups the applicant will incorporate into DCA Part 2, Tier 2, Section 3.7.1.1.1, Section 3.7.2.4.6, Table 3.7.2-33 (new), and Table 3.7.2-34 (new). The applicant also provided markups that show (1) updates to DCA Part 2, Tier 1, Table 5.0-1, and DCA Part 2, Tier 2, Table 2.0-1, "Site Design Parameters," to clarify 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, (2) the addition of the definition of the term "SSE" to DCA Part 2, Tier 1, Section 1.1, and (3) the addition of notes to 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, to clarify the basis of the design-basis seismic loads for applicable SSCs.

Based on its review, the staff finds the applicant's response and proposed markups acceptable because the applicant provided the information about the seismic SSI analysis cases used to establish the design-basis seismic demands for the RXB, CRB, and other key SSCs including the NPM, RBC, RFT, and bioshield. The staff also confirmed that the applicant incorporated the proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Therefore, **RAI 8935, Question 03.07.02-24**, is resolved and closed.

<u>Site-Specific Seismic Analysis by Combined License Applicant</u>

DCA Part 2, Tier 2, COL Item 3.7-5, addresses the need for a site-specific SSI analysis of the NuScale seismic Category I structures. However, the staff found COL Item 3.7-5 in DCA Revision 0 did not specify actions for the COL applicant to take based on the results of the site-specific SSI analysis. The staff found a similar issue with COL Item 3.7-6 in DCA Revision 0 that addresses the site-specific SSSI analysis. Therefore, the staff issued **RAI 8936, Question 03.07.02-12**, asking the applicant to expand COL Item 3.7-5 and COL Item 3.7-6 to ensure that the site-specific seismic demands of the standard design SSCs are bounded by the corresponding DCA demands and, if not bounded, to ensure the standard design SSCs are modified to accommodate the site-specific demands.

In its response dated August 30, 2017 (ADAMS Accession No. ML17242A281), to **RAI 8936, Question 03.07.02-12**, the applicant indicated that it will revise DCA Part 2, Tier 2, Section 3.7.2 and Table 1.8-2, by expanding COL Item 3.7-5 and COL Item 3.7-6 to ensure that the site-specific seismic demands of the standard design SSCs are bounded by the corresponding seismic demands in the DCA and, if not bounded, to ensure the standard design SSCs show appropriate margin or are appropriately modified to accommodate the site-specific demands. The applicant provided markups that will be incorporated into the DCA. The staff finds the applicant's RAI response to be acceptable because the applicant proposed to expand the affected COL Items to ensure that the site-specific seismic demands are bounded by the corresponding DCA seismic demands or take an appropriate action to accommodate the site-specific demands. The staff confirmed that the applicant incorporated its proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Therefore, **RAI 8936, Question 03.07.02-12**, is resolved and closed.

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 finds 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 are, therefore, acceptable.

<u>Effect of Structure-Soil-Structure Interaction on the Reactor Building and Control Building</u>

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.5.2, indicates that the ISRS from the triple building model are considered for the design of SSCs in the RXB. However, the applicant did not consider ISRS from the triple building model for the design of SSCs in the CRB. In DCA Part 2, Tier 2, Figures 3.7.2-106 and 3.7.2-107, the applicant presented the RXB ISRS for floor at elevations 24′ and 25′, respectively. The staff notes significant difference in the ISRS (both in shape and amplitude) for an elevation difference of only 1 foot. However, the applicant did not discuss a possible cause for such difference. Therefore, the staff issued **RAI 8935 Question 03.07.02-26**, asking the applicant to justify why it did not consider the ISRS from the triple building model for the design of SSCs in the CRB. The staff also asked the applicant to explain the noticeable difference in ISRS between floor elevations 24′ and 25′.

In its response dated November 29, 2018 (ADAMS Accession No. ML18333A359), and in the supplemental response dated January 28, 2019 (ADAMS Accession No. ML19028A234), to **RAI 8935 Question 03.07.02-26**, the applicant stated that it computed the ISRS in the CRB from the triple building model to account for the SSSI effects and identified substantial enhancements in the ISRS at several locations in the CRB from SSSI effects. The applicant further stated that the design-basis ISRS for equipment qualifications in the CRB are now established by enveloping the ISRS from the standalone CRB model for SSI effects and the triple building model for SSSI effects. The applicant also provided DCA markups that capture the findings discussed in the RAI response, which include updated design-basis ISRS that envelop the ISRS from the standalone CRB model and the triple building model. The staff finds the applicant's RAI response and the accompanying DCA markups to be acceptable because the design-basis ISRS for CRB equipment qualifications are conservatively established by enveloping the ISRS and accounting for both the SSI and SSSI effects.

The applicant stated that the ISRS in DCA Part 2, Tier 2, Figure 3.7.2-107 (elevation 24′), is an envelope of three nodal locations along the exterior wall on the north face, and these locations provide a good representation of the corridors surrounding the pool. However, the ISRS in DCA Part 2, Tier 2, Figure 3.7.2-108 (elevation 25′), is an envelope of 22 interior nodes in the pool region. The factors contributing to the noticeable difference in the ISRS include variations in basemat stiffness and dynamic response of the interior components (e.g., NPM, interior walls, pool water). The staff finds the applicant's explanation about the differences in ISRS at elevations 24′ and 25′ to be acceptable. Based on this review, the staff is tracking **RAI 8935 Question 03.07.02-26**, as **Confirmatory Item 03.07.02-7**, pending the update to the next revision of the DCA.

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, 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. The staff finds 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, 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 finds that, since the applicant does not use a method based on modal combination, no further review on 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 for assessing 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, "Interface Requirements—Site Specific Structures," 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, "Interaction of Non-Seismic Category I Structures with Seismic Category I Structures," the applicant described the criteria used to provide reasonable assurance that the failure of nonseismic Category I structures under the effect of a seismic event does not impair the integrity of an adjacent seismic Category I structure. 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.

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.8, provides information on the interaction of nonseismic Category I structures with seismic Category I structures. The applicant indicated that the upper portion of the CRB located above elevation 120′ 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. However, the applicant did not discuss potential seismic interactions of these buildings with adjacent seismic Category I structures, the RXB and CRB. Therefore, the staff issued **RAI 9036 Question 03.07.02-28**, asking the applicant to provide analysis or information that ensures that the failure of these nonseismic Category I structures will not impair the integrity of an adjacent seismic Category I structure during a design-basis seismic event.

In its response dated October 17, 2017 (ADAMS Accession No. ML17290B261), to **RAI 9036, Question 03.07.02-28**, the applicant indicated 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 applicant also noted that DCA Part 2, Tier 2, includes three COL information items (COL Items 3.3-1, 3.4-6, and 3.7-4) to ensure 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 nonseismic Category I structures will not impair the integrity of an adjacent seismic Category I structure. The staff confirmed that NuScale DCA Part 2, Tier 2, COL Items 3.3-1, 3.4-6, and 3.7-4, specify that a COL applicant that references the NuScale DC will confirm that nearby structures exposed to a site-specific tornado, hurricane, external flooding, or SSE will not collapse and adversely affect the adjacent seismic Category I structures.

Based on its review, the staff finds the applicant's response acceptable because the applicant's method of handling potential interaction of nonseismic Category I structures with seismic Category I SSCs is consistent with acceptance criteria of DSRS Section 3.7.2.II.8, and because it supports the information in DCA Part 2, Tier 1, Section 4.1, in that the turbine generator buildings, central utility buildings, and annex buildings are identified as site-specific structures not within the scope of the NuScale certified design, and their failure will not cause the failure of any of the seismic Category I structures within the scope of the NuScale certified design. Therefore, **RAI 9036, Question 3.07.02-28**, is resolved and closed.

COL Item 3.7-6 in DCA Part 2, Tier 2, Table 3.7.2-1, states that, "A COL applicant that references the NuScale Power Plant design certification will perform a SSSI analysis that includes the RXB, CRB, RWB and both Turbine Generator Buildings." However, Revision 0 of the DCA does not include an SSSI analysis involving the turbine generator buildings. Therefore, the staff issued **RAI 9036, Question 03.07.02-29**, asking the applicant to state (1) a justification for not including in the DCA the SSSI analysis involving the turbine generator buildings, (2) whether NuScale intends to provide in the DCA any guidelines on SSSI analysis involving the turbine generator buildings for a COL applicant to follow, and (3) whether the distance between the RXB and turbine generator buildings is considered a NuScale standard design parameter.

In its response dated October 17, 2017 (ADAMS Accession No. ML17290B261), to **RAI 9036, Question 03.07.02-29**, the applicant indicated that (1) the turbine generator buildings are site-specific buildings and are not included in the scope of the NuScale certified design; therefore, the DCA does not include an analysis involving the turbine generator buildings, (2) NuScale will not provide guidelines on the SSSI analysis of the turbine generator buildings, and the COL applicant will be required to fulfill COL Item 3.7-4 on the site-specific SSE analyses, and (3) the turbine generator buildings are located a minimum of 70 feet away from the RXB to reduce the potential impact on seismic Category I structures. The applicant provided markups for DCA Part 2, Tier 2, Section 3.8.4.1.4, that reflect the information discussed in item (3) above.

Based on its review, the staff finds the applicant's response and accompanying markups to be acceptable because the Turbine Generator Buildings are site-specific structures and the issue of seismic interaction between the Turbine Generator Buildings and seismic Category I SSCs will be handled by the COL applicant. The staff confirmed that the applicant incorporated its proposed markups into DCA Part 2, Tier 2, Section 3.8.4.1.4. Therefore, **RAI 9036, Question 03.07.02-29**, is resolved and closed.

3.7.2.4.9 Effects of Parameter Variations on Floor Response Spectra

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 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. 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. The response spectra broadening is performed in accordance with RG 1.122 and the applicant used stiffness reduction factors consistent with the criteria in ASCE 43-05, which are acceptable to the staff.

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

DCA Part 2, Tier 2, Revision 0, 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 seven NPMs and reported the results obtained from the study. However, the staff could not establish from the study that the sensitivity analyses performed would bound all the potential operating configurations with fewer than the full complement. Therefore, the staff issued **RAI 8936 Question 03.07.02-8**, asking the applicant to specify the potential operating configurations with less than 12 NPMs and demonstrate that, for the operating configurations considered, the design-basis demands presented in the DCA bound the seismic demands at critical building locations. The staff also requested that a COL information item be considered for a COL applicant to ensure that the COL applicant's specified operating configurations would be acceptable based on the site-specific analysis accounting for the site parameters.

In the response dated October 3,2017 (ADAMS Accession No. ML17276B886), to **RAI 8936 Question 03.07.02-8**, the applicant stated that the design of the NPMs and supporting components allows the plant to be operated at less than the full complement of 12 NPMs. The applicant also stated that in order 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.

The applicant further recognized that the analyzed scenarios may not account for all potential site-specific configurations. Therefore, in the markups provided as part of the RAI response, the applicant added a new COL Item 3.7-10 to DCA Part 2, Tier 2, Section 3.7.2.9.1.5, which requires that a COL applicant perform a site-specific configuration analysis that includes the RXB with the applicable configuration layout of the desired NPMs. Specifically, the COL applicant will confirm that the following quantities are bounded by the corresponding certified design seismic demands and, if not, the standard design will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands:

i. 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

ii. the maximum forces in the NPM lug restraints and skirts

iii. 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

iv. the maximum forces and moments in the east and west wing walls and pool walls (see DCA Part 2, Tier 2, Table 3.7.2-32)

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

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

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

viii. RXB crane wheels at elevation 145′ 6″: bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-114

ix. CRB east wall at elevation 76′ 6″: bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-119

x. CRB south wall at elevation 120′ 0″: bounded by ISRS in DCA Part 2, Tier 2, Figure 3.7.2-121

Based on its review, the staff finds the applicant's response acceptable because the applicant addressed the staff's concern through sensitivity studies on operating configurations at less than the full complement of 12 NPMs, and because the applicant provided a COL information item ensuring that site-specified 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. The staff confirmed that the applicant incorporated the proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Therefore, **RAI 8936 Question 03.07.02-8**, is resolved and closed.

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.

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.11, states that "to allow for the effects of accidental torsion, a 5 percent eccentricity is incorporated into the SASSI2010 element forces using SRSS." However, the staff notes Equation 3.7-17 in DCA Part 2, Tier 2, Section 3.7.2.11, indicates that the effect of accidental torsion is accounted for by increasing the maximum element forces by 5 percent, which is not consistent with guidance in DSRS Section 3.7.2.II.11. Therefore, the staff issued **RAI 8936 Question 03.07.02-9**, requesting the applicant to justify the use of the method represented by Equation 3.7-17.

In the response dated October 3, 2017 (ADAMS Accession No. ML17276B886), to **RAI 8936 Question 03.07.02-9**, the applicant stated that (1) the methodology chosen to account for accidental torsion was to increase the maximum horizontal element forces by 5 percent and combine them with the maximum vertical forces using SRSS, and (2) because torsion is the product of force and distance, increasing the seismic forces by 5 percent is equivalent to increasing the eccentricity by 5 percent. The applicant also provided markups for DCA Part 2, Tier 2, Section 3.7.2.11, stating that it had increased the element demand forces and moments obtained from SASSI2010 from E-W and N-S CSDRS (and CSDRS-HF) inputs by 5 percent to account for accidental torsion.

The staff reviewed the applicant's approach to addressing accidental torsion by increasing the horizontal element forces by 5 percent in lieu of increasing the eccentricity by 5 percent. The staff was unable to determine the validity of the applicant's approach, which involves a uniform increase of horizontal element forces by 5 percent in contrast to the DSRS methodology that involves an increase of 5 percent torsional eccentricity on a floor level. DSRS Section 3.7.2.II.11 specifies that to account for accidental torsion, an additional eccentricity of ±5 percent of the maximum building dimension should be assumed for both horizontal directions and that the magnitude and location of the two eccentricities are determined separately for each floor elevation. The staff also notes that similar guidance on accounting for accidental torsion is available in industry standards; for example, ASCE 4-16 specifies that the effect of accidental torsion be calculated at each floor level by static analysis, assuming a torsional moment equal to the product of the story shear and 5 percent of the plan dimension perpendicular to the direction of motion of the structure at that level.

The staff finds that the applicant has not demonstrated that an eccentricity of 5 percent of the building dimension is equivalent to a 5-percent increase in the elemental horizontal forces. Therefore, the staff closed **RAI 8936, Question 03.07.02-9,** as unresolved and is tracking the technical issue in a followup **RAI 9254, Question 03.07.02-33**, dated January 3, 2018. The staff requested the applicant to demonstrate that its method conforms to provisions in the DSRS or is more conservative. In its response dated March 2, 2018 (ADAMS Accession No. ML18061A063), to **RAI 9254, Question 03.07.02-33**, the applicant stated that the lateral load-resisting system of the RXB is the reinforced concrete shear walls, and the most efficient shear walls to resist the torsional effects are the exterior walls along the perimeter of the building. The applicant considered only these exterior walls to resist accidental torsion and conservatively ignored the resistance of the interior walls. The applicant then showed that this reduces the problem to a closed thin-wall section subject to torsional effects and demonstrated that the 5-percent amplification of horizontal element forces is conservatively equivalent to the 5-percent mass offset method of the DSRS for addressing accidental torsion.

Based on its review, the staff finds the applicant's response and proposed markups acceptable because they demonstrate that the applicant's methodology for addressing accidental torsion conservatively meets the intent of the acceptance criteria set forth in DSRS Section 3.7.2. The staff confirmed that the applicant incorporated the proposed markups into DCA Part 2, Tier 2, Section 3.7.2. Therefore, **RAI 9254, Question 03.07.02-33**, is resolved and closed.

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, in accordance with DSRS Section 3.7.2.II.12, 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.

DCA Part 2, Tier 2, Revision 0, Section 3.7.2.15, refers to DCA Part 2, Tier 2, Revision 0, Section 3.7.1.3, for the damping ratios used for seismic analysis of the RXB and CRB. However, DCA Part 2, Tier 2, Section 3.7.1.3, does not provide information on analysis procedure for damping—either the composite modal damping approach or the modal synthesis technique. Therefore, the staff issued **RAI 8936, Question 03.07.02-11**, asking the applicant to provide information on the analysis procedure for damping used in various seismic analyses for NuScale seismic Category I SSCs.

In its response dated October 3, 2017 (ADAMS Accession No. ML17276B886), to **RAI 8936, Question 03.07.02-11**, 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 suitable for the type of analysis performed are followed. Dynamic fluid-structure interaction analysis of the RXB and stability analysis of the RXB and CRB with ANSYS use Rayleigh damping; SSI analysis of the RXB and CRB with SASSI2010 uses hysteretic damping. 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 the implementation of damping in the dynamic analysis of the seismic Category I structures and finds 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 that can be represented by a series of harmonic loads.[1] In SASSI, the transient input motion is decomposed into a series of harmonic motions through the Fourier Transform.

Based on its review, the staff finds the applicant's use of Rayleigh damping in ANSYS and hysteretic damping in the SASSI analysis acceptable. The staff also finds the DCA markups provided by the applicant adequately captured the information in the RAI response. The staff confirmed that the applicant incorporated the proposed markups in DCA Part 2, Tier 2, Section 3.7.2. Therefore, **RAI 8936, Question 03.07.02-11**, is resolved and closed.

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

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[1] S.L. Kramer, Geotechnical Earthquake Engineering, Prentice Hall, 1996.

walls, and soil frictional forces for seismic Category I structures. In DCA Part 2, Tier 2, Section 3.7.2.12, 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 finds 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

COL 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.	3.7.2.8
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.	3.7.2.16
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 design SSCs will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands.	3.7.2.16

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 EL 75′-0″: bounded by in-structure response spectra in Figure 3.7.2-110 • RXB west exterior wall at EL 126′-0″: bounded by in-structure response spectra in Figure 3.7.2-112 • RXB crane wheels at EL 145′-6″: bounded by in-structure response spectra in Figure 3.7.2-114 • CRB east wall at EL 76′-6″: bounded by in-structure response spectra in Figure 3.7.2-119 • CRB south wall at EL 120′-0″: bounded by in-structure response spectra in Figure 3.7.2-121 If not, the standard design will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands.	3.7.2.9
COL Item 3.7-11	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific analysis that, if applicable, assesses the effects of soil separation. The COL applicant will	3.7.2.1

	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.	

The staff reviewed the COL information items 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, except for the Open Items identified above that will be dispositioned based on the review of supplemental information to be provided by the applicant..

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Introduction

DCA Part 2, Tier 2, Revision 0, 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; concrete dams; 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, "Dynamic Testing and Analysis of Systems, Structures, and Components," Revision 4, issued March 2017 and SRP Section 3.9.3, "ASME BPV Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures" Revision 3, 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:

A. seismic analysis methods
B. determination of the number of earthquake cycles
C. procedure used for analytical modeling
D. basis for selection of frequencies
E. analysis procedure for damping
F. three components of design ground motion
G. combination of modal responses
H. interaction of nonseismic Category I subsystems with seismic Category I SSCs
I. multiply supported equipment and components with distinct inputs
J. use of equivalent vertical static factors
K. torsional effects of eccentric masses
L. seismic Category I buried piping, conduits, and tunnels
M. methods for seismic analysis of seismic Category I concrete dams
N. methods for seismic analysis of aboveground tanks

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) 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, “Seismic System Analysis.” 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, “Seismic Design Parameters,” 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 TS 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:

- 10 CFR Part 50, Appendix A, 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. 10 CFR Part 50, Appendix S, 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, 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

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 Section3.7.2.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 SRP 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 500,000 kips. A subsystem can be decoupled if the weight is less 5,000 kips. The larger subsystems, the NPM and RBC, weigh approximately 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 Revision 2, Tier 2, Section 9.1.5, discusses the RBC, and is evaluated in SER Chapter 9.

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 staff evaluates the fuel storage racks in SER 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 125′. The bioshields provide radiation protection in the RXB and support personnel access. The bioshields are attached to the bay walls and outer pool wall. A vertical assembly is attached to the horizontal slab. 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. Staff's evaluation of the Bioshield is discussed below.

Bioshield

In response to **RAI 9447, Question 03.11-19**, (ADAMS Accession No. ML18320A254) the applicant stated that the horizontal assembly of the bioshield consists of a 23.5-inch-thick, 5,000 psi reinforced concrete horizontal slab. The vertical assembly is constructed of a stainless steel tube steel framing system and series of radiation panels. The vertical assembly is vented for heat removal during normal operations as well as heat and pressure removal during a HELB. Furthermore, the applicant developed ISRS for multiple locations in the RXB and from that selected two nodes to design the bioshield. The calculated natural frequency of the bioshield is used for determining the maximum acceleration in all three directions for use in the seismic design.

In **RAI 8838, Question 03.08.04-1**, the staff requested the applicant to describe the mechanism for restraining a bioshield mounted on an adjacent bioshield during refueling operations. The staff also requested the applicant to provide analysis and design criteria to ensure no adverse interactions occur between the seismic Category II bioshield and the adjacent seismic Category I structures during refueling operations.

The staff reviewed the applicant's response to these RAIs and the pertinent documents related to the structural design of the bioshield during an audit (ADAMS Accession No. ML19098A162) at the applicant's facility. During this review, the staff identified the following issues that needed further evaluation:

- NuScale employed an equivalent static approach for seismic analysis of the horizontal bioshield. However, the analysis did not include a factor of 1.5 applied to the peak spectral acceleration as required in DCA Revision 2, Part 2, Tier 2, Section 3.7.3.1.2.

- NuScale used the zero period acceleration in the horizontal directions, assuming the horizontal slab is rigid. However, the applicant did not provide the technical basis for this assumption.

- Concerning the FEM approaches to model the stacked configuration, the staff stated that representing only the mass of the upper bioshield in a stacked configuration may not adequately represent the dynamic modal characteristics of the system. The applicant would need to provide further justification for the qualification of the bioshield, including the anchors.

- The vertical panel displacements should be checked for potential overstress in members within the panels and weldment.

- In the next revision of the DCA, the applicant should address the design code and seismic Category II and I requirements for the bioshield design, including the anchor bolts, hinge pin, and lug, as they are considered mounting for the bioshield.

- The applicant should provide D/C ratios in the next revision of the DCA.

The staff is tracking these issues in **RAI 9447, Question 03.11-19**. The staff is waiting for the applicant's supplementary response to **RAI 9447** related to the design of the bioshield; therefore, **RAI 9447, Question 03.11-19**, is **Open Item 03.07.03-1**.

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 of Critical Damping 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

DCA Part 2, Tier 2, Section 3.7.3.6, "Three Components of Earthquake Motion," indicates 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, "Three Components of Earthquake Motion." 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

DCA Part 2, Tier 2, Section 3.7.3.7, "Combination of Modal Responses," indicates 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 1 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, 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 method and the independent support motion method may be used to address multiply supported equipment and components. For the independent support motion 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 3.9.2.2.8 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 soil structure interaction 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 hertz, 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

There are no COL information items for this area of review.

3.7.3.6 Conclusion

The staff will complete the review and make findings when it receives and evaluates responses to the Open Items discussed above.

3.7.4 Seismic Instrumentation

3.7.4.1 Introduction

This SER section presents the instrumentation system for measuring the effects of an earthquake.

3.7.4.2 Summary of Application

DCA Part 2, Tier 1:

DCA Part 2, Tier 2: 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.

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

Technical Specifications: When an earthquake occurs, a seismic monitoring system (SMS) records the ground-motion data. The SMS includes the sensors, wiring between the sensors, the control cabinet, and the instrumentation in the control cabinet. The controller processes the data and provides alarm notification to the main control room via the plant control system. Because the plant control system is not a seismic Category I system, additional seismic Category I annunciation equipment is located in the main control room to alert operators of a seismic event. This annunciation is part of the SMS.

Seismic sensors will be located in the free field, RXB, and 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.

DCA Part 2, Tier 2, Section 3.7.4.1, states that the NuScale design requires a deviation from the guidance in RG 1.12, "Nuclear Power Plant Instrumentation for Earthquakes," Revision 3, 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.

In the selection of the exact sensor locations, the COL applicant shall 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 24′ 0″. 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 75′ 0″ 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 50′ 0″, near the intersection of gridlines CB-4 and CB-A.

- An eighth sensor unit is located in the CRB at elevation 100′ 0″ on the east face of gridline CB-1 between CB-B and CB-C.

DCA Part 2, Tier 2, Section 3.7.4.3, states that the 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.

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

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

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

Technical Specifications: There are no TS 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 3, issued November 2017
- RG 1.166, Revision 0

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 its original DCA submittal, the applicant followed the older version of RG 1.12 (Revision 2), issued March 1997. The staff agreed that the locations of the free-field seismic sensors are site specific and the COL applicant should discuss them. However, because NuScale proposes to deviate from the RG 1.12 recommendations, the staff requested that NuScale prescribe the locations of the structural seismic instrumentation in DCA Part 2.

The staff issued **RAI 8927,** dated August 5, 2017 (ADAMS Accession No. ML17217A013), requesting the applicant to specify instructural locations of the instruments. In its response dated September 29, 2017 (ADAMS Accession No. ML17272A607), the applicant specified locations of the instructural sensors within the RXB and CRB and updated DCA Part 2, Tier 2, Section 3.7.4.2. However, the staff found the SMS proposed by the applicant not fully adequate to assess the response of the plant to an earthquake.

The staff issued **RAI 9228**, dated December 15, 2017 (ADAMS Accession No. ML17349A328), requesting an additional free-field downhole seismic instrument at the elevation corresponding to the GMRS or FIRS and also an additional seismic instrument on the basemat to record the rocking or torsional responses predicted by the engineering analyses. In its response dated February 12, 2018 (ADAMS Accession No. ML18043B166), the applicant 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. The applicant also updated DCA Part 2, Tier 2, Section 3.7.4.2.

The updated DCA Part 2 used the most recent version of RG 1.12 (Revision 3), and provided description of eight seismic sensor locations ensuring that the site, RXB, and CRB are adequately instrumented for a seismic event.

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

COL 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 Post-earthquake Actions,” Revision 0 (or later). This information is to be provided as noted below.	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.	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 following the most recent RG 1.12 (Revision 3). 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 is an integral portion of the NPM, with the following primary functions:

- Provide an essentially leak-tight barrier to contain fission product releases for the RCPB during design-basis events.
- 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.

DCA Part 2, Tier 2, Section 3.8.2, provides 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 inservice inspection programs

3.8.2.2 Summary of Application

DCA Part 2, Tier 1: 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 containment vessel.

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.

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 3.6, "Containment Systems."

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

- TR-0716-50424-P, Revision 0, "Combustible Gas Control" dated December 31, 2016

- TR-0916-51502-P, Revision 1, "NuScale Power Module Seismic Analysis" dated September 28, 2018

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 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, GDC 16, “Containment Design,” as it relates to the ability of the reactor containment to act as an essentially leak-tight barrier to prevent the uncontrolled release of radioactive effluents to the environment

- 10 CFR Part 50, Appendix A, 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.

- 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, “Control of Combustible Gas Concentrations in Containment,” Revision 3, issued March 2007

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

- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” issued June 2007

- RG 1.216, “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 RCPB 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, Sections 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 DSRS Section 3.8.2.I.1.A for geometry of the containment vessel, including sketches showing plan views at various elevations and sections in at least two orthogonal directions. However, the two figures referenced in DCA Part 2, Tier 2, Section 3.8.2.1, Figures 3.8.2-1 and 6.2-1, do not provide this detail. The drawings should indicate ASME Class or QG boundaries, and some drawings lack measurements. For example, the figures should show boundaries between the CNV and the RPV (RPV support and support ledge). The staff issued **RAI 8858, Question 03.08.02-1**, dated September 1, 2017 (ADAMS Accession No. ML17244A861), asking the applicant to provide the various sketches showing plan views as described in DSRS Section 3.8.2.I.1.A and in accordance with 10 CFR 52.47, "Contents of Applications; Technical Information," which requires the DC applicant to include a description and analysis of the SSCs sufficient to permit understanding of the system designs.

In response to **RAI 8858, Question 03.08.02-1** (ADAMS Accession No. ML17300B428), the applicant added dimensions to 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. The applicant added Figures 3.8.2-8 and 3.8.2-9 to show views of the CNV–RPV boundary at the supports.

In response to followup **RAI 9362, Question 03.08.02-15**, dated December 10, 2017 (ADAMS Accession No. ML18344A512), the applicant added Figure 3.8.2-10 to DCA Part 2, Tier 2, Section 3.8.2, to identify 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 (1/2-inch 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 the added Figures 3.8.2-8, 3.8.2-9, and 3.8.2-10. The applicant also revised DCA Part 2, Tier 2, Sections 5.4.2.4 and 6.1.1.2, stating that no socket welds are used on lines greater than or equal to 3/4-inch NPS and that socket welds less than 3/4-inch NPS conform to 10 CFR 50.55a(b)(1)(ii).

The staff finds the response to **RAI 9362, Question 03.08.02-15**, acceptable because it provides the level of detail specified in DSRS Section 3.8.2.I.1.A and and is in accordance with 10 CFR 50.55a(b)(1)(ii).

The staff is tracking **RAI 9362, Question 03.08.02-15**, as **Confirmatory Item 03.08.02-1**, pending the incorporation of revisions to DCA Part 2, Tier 2, Sections 5.4.2.4 and 6.1.1.2.

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, 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 ½ through NPS 24 Metric/Inch Standard", are rated to be acceptable for the CNV design pressure of 1,000 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, "NuScale Power Module Mechanical Equipment." The staff finds this acceptable in 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.I.2.

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 0, 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 description is insufficient, and the report does not provide the seismic and LOCA stress results. The staff issued **RAI 8858, Question 03.08.02-3**, dated September 1, 2017 (ADAMS Accession No. ML17244A861), asking the applicant to provide the seismic analysis details and stress results under Service Level D condition for the CNV, which should include a description of structure modeling, input motion (time history or in-structure response spectrum), major assumptions, acceptance criteria, fluid structural interaction considerations, mass distribution, damping values, dominant frequency and modes shape plots, gap/impact modeling, and seismic and LOCA stress results.

In response to followup **RAI 9362, Question 03.08.02-16**, dated May 29, 2018 (ADAMS Accession No. ML18149A651), the applicant sent a letter dated February 4, 2019 (ADAMS Accession No. ML19035A682), committing to make the updated CNV stress document with the new seismic loads available for audit.

RAI 9362, Question 03.08.02-16 will be tracked as **Open Item 03.08.02-1**, pending the audit of the updated CNV stress document with the new seismic loads.

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.I.3.

The staff reviewed TR-0716-50424-P, Section 3.3.4.4. The staff found that the containment vessel shell is within ASME Service Level C limits for loads from 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.

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 finite element analysis 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," dated December 2017 (ADAMS Accession No. ML17348A527), addresses the details of the predicted containment internal pressure capacity above design pressure.

In **RAI 9459, Question 03.08.02-18** (ADAMS Accession No. ML18260A373, response dated September 17, 2018), the staff requested NuScale to provide details on how the CNV RPV support meets the requirements in ASME BPV Code, Section III, Subsection NF, and the effect of the RPV support on the containment stresses. In response, the applicant stated that it has performed a preliminary evaluation, calculating the Level D stresses, to size the CNV RPV support and to confirm that the Level D limits can be met with the current support design. In its letter dated February 4, 2019, the applicant stated it will make the primary stress calculation for the CNV RPV support available for audit in July 2019.

RAI 9459, Question 03.08.02-18, will be tracked as **Open Item 03.08.02-2**, pending the audit of the primary stress calculations for the CNV RPV support.

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). As indicated in its letter dated February 4, 2019, the applicant will make the fatigue evaluation of the limiting locations for the CNV available for audit in July 2019.

The staff is tracking the fatigue evaluation of the CNV as **Open Item 03.08.02-3**, until reviewed in a future audit.

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. ASME BPV CodeDCA 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-5000ASME BPV Code. 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 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 inservice inspection 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.I.7. SER Section 6.2.1.6 gives the staff evaluation of the inservice inspection of the CNV.

3.8.2.5 Combined License Information Items

There are no COL information items for this area of review.

3.8.2.6 Conclusion

Because of the open items identified above, the staff is unable to make a safety finding at this time. 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 regulatory requirements delineated in Section 3.8.2.3 of this report are satisfied, except for the Open Items identified above that will be addressed by an audit to be performed at a future date.

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 containment vessel 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 RWB (seismic Category II structure), 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 criterial; materials, quality control, and special construction techniques; and testing and inservice inspection 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 34 feet east of the RXB. A below-grade tunnel extends out from the CRB to the RXB. There is a 6-inch expansion gap between the end of the tunnel and the RXB walls. The RWB sits on a separate foundation, approximately 25 feet west of the RXB (i.e., at the end opposite from the CRB). 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: 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.

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

Technical Reports: There are no TRs for this review.

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 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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 3.8.4 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, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants” Revision 1, issued May 2009

- RG 1.91, “Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants” Revision 2, issued April 2013

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

- RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)” Revision 2, issued November 2001

- RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants” Revision 2, issued November 2001

- RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” Revision 3, 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. 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 86 feet below grade level, supported on a single basemat foundation. It extends approximately 81 feet above grade level for a total overall height of approximately 167 feet 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 5 feet 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 24′, to their connection with the roof at elevation 163′. 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 5 feet wide and extend 5 feet out from the wall. The corner pilasters are 12.5 feet wide and extend 2.5 feet out from the wall. The RXB is 346 feet wide (excluding pilasters) in the E-W direction and 150.5 feet wide (excluding pilasters) in the N-S direction and is centered on a below-grade 358-foot by 162.5-foot 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 34 feet to the east and embedded approximately 55 feet below grade. It extends approximately 41 feet above grade level, for a total height of approximately 96 feet 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 3 feet wide and extend 3 feet out from the wall. In addition, there are four corner pilasters, 7.5 feet wide and extending 1.5 feet out from the wall. The CRB is 81 feet wide (excluding pilasters) in the E-W direction and 119 feet, 8 inches wide (excluding pilasters) in the N-S direction. It is centered on a below-grade 91-foot by 129-foot, 8-inch basemat. The CRB includes a below-grade tunnel that extends from the CRB to the RXB. There is a 6-inch 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 SSC on the upper steel structure has no safety-related function and is 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.

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 specification applicable to the 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 exceptions. 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 containment vessel. Further, as stated in DCA Part 2, Tier 2, Table 1.9-2, RG 1.127 is not applicable to the standard design structures. 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, DCA Part 2, Tier 2, Table 1.9-2 and Section 3.2.1.4, both address conformance with RG 1.143.

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, "Specification for Safety-Related Steel Structures for Nuclear Facilities," 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 that reasonably covers 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, DCA Part 2, Tier 2, Sections 3.8.4.3.1 through 3.8.4.3.22, discuss the loads, and DCA Part 2, Tier 2, 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.

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 (To) and accident thermal loads (Ta), respectively. The staff issued **RAI 8971, Question 03.08.04-13**, dated August 5, 2017 (ADAMS Accession No. ML17217A003), to seek additional information and clarification of the applicant’s approach for addressing operating and accident thermal loads and the applicant’s consideration of the controlling load combination(s) in its design evaluation. The staff is awaiting a supplemental response. Accordingly, the staff is tracking **RAI 8971, Question 03.08.04-13**, as **Open Item 03.08.04-1**, pending the applicant’s response.

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 (Yj), pipe break reaction loads (Yr), and missile impact loads (Ym), respectively. The applicant determined the Yj and Yr loads based on its pipe rupture hazard analysis described in TR-0818-61384-P. SER Section 3.6 evaluates the applicant’s pipe rupture hazards analysis. 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 Yj and Yr loads to be bounded by the RXB capacity. With respect to Ym 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, staff’s review and the site-specific verifications to be performed by the COL applicant, the staff finds the applicant’s consideration of Yj, Yr, and Ym 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.

<u>Analysis Procedures</u>

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 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. 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 envelop 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 3.7.2.1.2.2, 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 3.7.2.1.2.4, 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.

<u>Design Procedures</u>

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. 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. In general, 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. The staff finds the aforementioned determination of the design-basis demands, which envelops analysis results considering a range of key structural and site parameters, to be conservative and acceptable.

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 is awaiting a supplemental response to **RAI 8975, Question 03.08.04-26**, with updates to DCA Part 2, Tier 2, to include structural capacity tables examined during the seismic/structural audit. The staff is tracking this audit action item in the NuScale closure plan for RAIs related to DCA Part 2, Tier 2, Sections 3.7 and 3.8, and it is **Open Item 03.08.04-2** in this SER.) 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 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). 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. The following material discusses the critical sections for which the staff need additional information or clarification.

Reactor Building Roof

The RXB roof is a 4-foot-thick slab composed of a flat section at elevation 181′ between gridlines A.7 and D.3 and two inclined roof sections, sloping down from elevation 181′ to elevation 163′ 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. 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.

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 found 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 (ADAMS Accession No. ML19098A162), 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 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.

Wall at Gridline 1

The wall at gridline 1 is a 5-foot-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 (ADAMS Accession No. ML19098A162), 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 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 RFT stand. The discussion below gives the staff's review of the RXB pool liner and aforementioned structural supports.

Reactor Building Pool Liner

DCA Part 2, Tier 2, Table 3.2-1, categorizes the RXB pool liner as a seismic Category I component. 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. During the seismic/structural audit in December 2018 (ADAMS Accession No. ML19098A162), the staff examined the applicant's design report for the RXB pool liner. Based on its examination, the staff requested the applicant to provide additional information in DCA Part 2, Tier 2, Section 3.8.4, with respect to the design the RXB pool liner. NuScale's closure plan for DCA Part 2, Tier 2, Sections 3.7 and 3.8, addresses the DCA Part 2, Tier 2, update with respect to above request for information. The staff is awaiting the supplemental information on the pool liner and tracking this as **Open Item 03.08.04-3**.

Reactor Building Cooling and Fuel-Handling Machine Rails

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 the American Society of Mechanical Engineers, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," NOG-1, issued 2004 (ASME NOG-1) criteria; 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.

Reactor Flange Tool Stand

DCA Part 2, Tier 2, Section 9.1.5.2.3, describes refueling operations, including the use of an RFT for de-tensioning and tensioning the reactor flange closure bolts and as a structural support during refueling operations. As stated in DCA Part 2, Tier 2, Table 3.2-1, the RFT stand is categorized as seismic Category I. The staff's review focused on evaluating the structural integrity of the RFT stand when subjected to the design-basis demands. The staff issued **RAI 8838, Question 03.08.04-1**, dated May 26, 2017 (ADAMS Accession No. ML17146B358), seeking a discussion of structural design information for the RFT stand because it was lacking in the application. The staff is awaiting the applicant's response. Accordingly, the staff is tracking **RAI 8838, Question 3.08.04-1**, as **Open Item 03.08.04-4**, pending the applicant's response.

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 25 feet 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.

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.

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 American Society for Testing and Materials (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 is 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 additional 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.

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 is calculated by ACI 349-specified formulas. Welded wire fabric for concrete reinforcement conforms to ASTM A185 (plain wire) or ASTM A497 (deformed wire).

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 (AWS) 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 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 within the scope of the primary design codes. Therefore, the staff finds these material specifications to be acceptable.

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 100′, 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, 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 inservice inspection 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.

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, 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

COL 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 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.	3.8.4.8

For COL Item 3.8-2, DCA Part 2, Tier 2, Section 3.8.4.8, provides criteria related to the comparison of analysis parameters to establish the acceptability of a standard design for a potential site. The criteria in DCA Part 2, Tier 2, Section 3.8.4.8, identified locations within the building and respective ISRS that the COL applicant will use to compare with its site-specific ISRS. The staff issued **RAI 8974, Question 03.08.04-23**, dated August 5, 2017 (ADAMS Accession No. ML17217A022), asking the applicant to augment the comparison criteria to address the additional analysis parameters in DSRS Section 3.7.1.II.4.A.viii. The staff is awaiting the applicant’s response. Accordingly, the staff is tracking **RAI 8974, Question 03.08.04-23,** as **Open Item 03.08.04-5**, pending the applicant’s response.

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 structure 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, except for the Open Items identified above that will be dispositioned based on the review of supplemental information to be provided by the applicant.

3.8.5 Foundations

3.8.5.1 Introduction

DCA Part 2, Tier 2, Section 3.8.5, “Foundations,” describes the seismic Category I structures of the RXB and CRB basemats.

3.8.5.2 Summary of Application

DCA Part 2, Tier 2, Section 3.8.5, addresses the following:

- description of foundations
- applicable codes, standards, and specifications
- loads and load combinations
- evaluation criteria for stability analysis
- results compared with structural acceptance criteria
- materials, quality control, and special construction techniques
- testing and inservice surveillance requirements

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Revision 0, Section 3.11, “Reactor Building,” and Section 3.13, “Control Building,” present information associated with this section.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.8.5, includes information on the physical layout, design, construction, and inspection.

ITAAC: DCA Part 2, Tier 1, Table 3.11-2, “Reactor Building ITAAC,” and Table 3.13-1, “Control Building ITAAC,” provide ITAAC items for seismic Category I structures.

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:

- 10 CFR Part 50, Appendix A, GDC 1, as they relate to safety-related structures being 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, 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

- 10 CFR Part 50, Appendix A 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 1.127, Revision 1, issued March 1978
- RG 1.142
- RG 1.160
- RG 1.206

3.8.5.4 Technical Evaluation

The staff review ensures that the foundation design meets the applicable requirements in GDC 1, 2, 4, and 5; and 10 CFR Part 50, Appendix B, are in accordance with DSRS Acceptance Criteria 3.8.5.II.

The staff reviewed DCA Part 2, Tier 2, Section 3.8.5, against the agency's regulatory guidance to ensure that DCA Part 2 represents the complete scope of information relating to this review topic. SRP Section 3.8.5 identifies seven specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations listed in DSRS Section 3.8.5.II and included in SER Sections 3.8.5(D)(a)(g). This section evaluates DCA Part 2, Tier 2, Section 3.8.5, for each of these seven SRP acceptance criteria.

DSRS Section 3.8.5 provides guidelines for the staff to use in reviewing the technical areas related to the design of foundations based on the requirements of GDC 1, 2, 4, and 5; 10 CFR Part 50, Appendix B. These technical areas include a description of the foundations; applicable codes, standards, and specifications; loads and load combinations; evaluated criteria for stability analysis; results compared with structural acceptance criteria; materials, quality control, and special construction techniques; and testing and inservice surveillance requirements.

The staff reviewed DCA Part 2, Tier 2, Section 3.8.5, in accordance with SRP Section 3.8.5 and related RGs and industry standards. In particular, the staff focused on the analysis and design of the NuScale foundations, emphasizing (1) material, (2) geometry, (3) codes and standards, (4) loadings and load combinations, and (5) design and analysis procedures.

Based on the response to **RAI 8963, Question 03.08.05-14**, (ADAMS Accession No. ML17290B267) dated October 17, 2017 and (ADAMS Accession No. ML18094B106) dated April 4, 2018, the applicant added COL Item 3.8-3 in DCA Part 2, Tier 2, Revision 2, Section 3.8.5.4.1.4, “Control Building Basemat Nonlinear Analysis Model Description,” 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.

The staff based its evaluation on a complete review of DCA Part 2, Tier 2, Sections 3.8.5.1 through 3.8.5.6, including all subsections.

This section describes the staff’s review and evaluation of DCA Part 2, Tier 2, Revision 2, Section 3.8.5, against the guidance of SRP Section 3.8.5 to ensure the applicant has adequately addressed identified technical issues.

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 Criteria 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 panthers of the RXB and CRN basemats as follows.

<u>Reactor Building Foundation</u>

The RXB basemat dimensions are 359 feet (109.42 meters) in the E-W direction, 163 feet (49.68 meters) in the N-S direction, with a minimum thickness of 10 feet (3.05 meters). The top of concrete elevations of the foundation, the refueling pool area, the elevator area and the sumps area are elevations 24′ (7.32 meters), 19′ (5.79 meters), 17′ (5.18 meters) and 20′ (6.10 meters), respectively.

The steel reinforcing bars at the perimeter of the RXB basemat extend 15 feet (4.57 meters) from the centerlines of the exterior wall and consist of six layers of #11 bars centered at 12-inches (30.48 centimeters) each way (N-S and E-W) at top and bottom surfaces, with two-legged stirrups of #6 bars centered at 12 inches each way.

The steel reinforcing bars at the interior sections of the basemat consist of four layers of #11 bars centered at 12 inches (30.48 centimeters) each way (N-S and E-W) at top and bottom surfaces, with one-legged stirrups of #6 bars centered at 12 inches each way.

<u>Control Building Foundation</u>

The CRB basemat dimensions are 130 feet (39.62 meters) in the E-W direction, 91 feet (27.74 meters) in N-S direction, with a thickness of 5 feet (1.524 meters).

The CRB reinforcement pattern at the perimeter of the basemat consists of four layers of #11 bars centered at 12 inches (30.48 centimeters) each way (N-W and E-W) at top and bottom, with two-legged stirrups of #6 bars centered at 12 inches (30.48 centimeters) each way, and at the center regions are three layers of #11 bars centered at 12 inch each way at top and bottom with two-legged stirrups of #6 bars centered at 12 inches each way.

The staff reviewed the descriptions of the foundations for RXB and CRB buildings to ensure 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 conducted a Phase 2 regulatory audit from December 3–7, 2018 (ADAMS Accession No. ML18324A551), to review the applicant's documents and to obtain additional information of the NuScale design of the seismic Category I structures. During the audit, the staff also reviewed the following structural concrete drawings to ensure 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 the thicknesses of the basemats in DCA Part 2, Tier 2, Revision 2, are as shown in the structural concrete drawings listed above. During the audit, the applicant described the rationale for excluding geometric discontinuations (vertical 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 conclude that the consideration of flat-bottom basemats in the analytical analyses is acceptable.

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.

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 3.8.5.3, describes loads and load combinations for the design and construction of the RXB and CRB. 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 309,495 kips (140,384 metric tons) and 40,500 kips (18,370 mt), 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 conducted a Phase 2 regulatory audit from December 3rd to 7th, 2018 (ADAMS Accession No. ML18324A551), to review the applicant's related documents and to obtain additional information on the NuScale design of the seismic Category I structures. During the audit, the staff 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. However, DCA Part 2, Tier 2, does not provide any information related to the requirement of 600 pounds per square foot (psf) (28.72 kilopascal (kPa)) for this construction load of concrete pour pressure, based on ACI 347, "Recommended Practice for Concrete Formwork." Therefore, the staff requested the applicant to describe this loading condition in DCA Part 2, Tier 2, and the applicant agreed. The staff also resorted to this calculation to address the applicant's response to **RAI 8963, Question 03.08.05-23**, below.

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 causes 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 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 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.

3.8.5.4.3.1 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 130 pound-force per cubic foot (2,082 kilograms per cubic meter) 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.

DCA Part 2, Tier 1, Table 5.0-1 and Table 2.0-1, “Site Design 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 0.5 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 0.5 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.

DCA Part 2, Tier 2, Tables 3.8.5-1 and 3.8.5-9, tabulate the surcharge load (σ_h) as 0.250 kilopound per square foot (ksf) (0.012 megapascal (MPa)) and uses the surcharge load in the design calculations of 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 surcharged loads are to be different for the RXB walls because of the effect of adjacent buildings (e.g., turbine generator buildings, RWB). Thus, the technical basis should provide for the 0.250 ksf (0.012 MPa) surcharge used in calculating the static lateral soil pressure.

Therefore, the staff issued **RAI 8963, Question 03.08.05-7**, asking the applicant to provide the basis for the surcharge pressure of 0.25 ksf (0.012 MPa), and why it was applied uniformly around the perimeter of the RXB and CRB embedded walls.

The applicant submitted the response and supplemental response dated October 17, 2017, and March 15, 2018 (ADAMS Accession Nos. ML17290B267 and ML18074A257, respectively),.to **RAI 8963, Question 03.08.05-7**. As discussed in a public meeting on January 30, 2018, the applicant provided a supplement to the original response to **RAI 8963, Question 03.08.05-7**, to address the surcharge pressures from the effects of adjacent buildings. The applicant considered the pressures on the walls of the RXB and CRB from the effects of adjacent buildings as part of the overall RXB and CRB static and dynamic analyses. The 0.25 ksf (0.012 MPa) surcharge pressure at the wall surface is in addition to the other surcharge loadings resulting from the SASSI2010 SSI analysis.

Based on its review, the staff finds the response to **RAI 8963, Question 03.08.05-7**, acceptable because the applicant described that the 0.25 ksf (0.012 MPa) surcharge pressure at the wall surface is in addition to the other surcharge loadings resulting from the SASSI2010 SSI analysis. Therefore, **RAI 8963, Question 03.08.05-7**, is resolved and closed.

DCA Part 2, Tier 2, Section 3.8.5.3.1, calculates the lateral soil forces for RXB and CRB. The forces for the RXB walls are calculated as 46,967 kips (21,303 mt) for the E-W wall and 21,429 kips (9.720 mt) for the N-S wall. The staff performed an independent check of the calculations for the CRB and determined the applicant calculated the forces correctly. DCA Part 2, Tier 2, Revision 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) = 345,847 kips (156,873 mt), F_Y (N-S) = 285,248 kips (129,386 mt), and F_Z (vertical) = 267,641 kips (121,399 mt).

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.

3.8.5.4.3.2 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 with 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.

3.8.5.4.3.3 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 307,702 kips (139,571 mt) and 5,274 kips (2,392 mt), respectively.

Based on the review, the staff found with 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 met DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.3.4 Design and Analysis Procedures

The staff reviewed the design and analysis procedure used for the seismic Category I 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.4.

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. Based on the review, the staff was not able to find sufficient information to make a safety assessment for the design of RXB basemat. Therefore, the staff issued **RAI 8963, Question 03.08.05-6,** asking the applicant to provide (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 as discussed in DSRS Section 3.8.5.II.4.N, and (4) settlement evaluations and figures showing reinforcement patterns for the RXB basemat.

The applicant submitted its response to **RAI 8963, Question 03.08.05-6**, in a letter dated December 20, 2018 (ADAMS Accession No. ML18354B330). The staff is currently reviewing the applicant's response, which has additional information related to basemat design and settlements.

DSRS Section 3.8.5 provides review guidance on maximizing the bending moments used in the design of foundations. Based on the staff's review, the applicant did not provide any discussion related to "stiff and soft spots" in the foundation soil. Therefore, the staff issued **RAI 8963, Question 03.08.05-14**, asking the applicant to describe how it is considering the "stiff and soft spots" in the basemat evaluations of seismic Category I structures.

The applicant submitted its response and supplemental responses dated October 17, 2017, February 21, 2018, and April 4, 2018 (ADAMS Accession Nos. ML17290B267, ML18052B566, and ML18094B106, respectively), to **RAI 8963, Question 03.08.05-14**. During a public meeting on February 14, 2018, the applicant provided the supplement response to **RAI 8963, Question 03.08.05-14**, to address how the basemat evaluations considered "stiff and soft spots." The applicant revised and provided markups for COL Item 3.8-3 in DCA Part 2, Tier 2, Revision 2, Section 3.8.5.4, "Design and Analysis Procedures," and in Table 1.8-2, stating that a COL applicant referencing 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.

Based on its review, the staff finds the applicant's response to **RAI 8963, Question 03.08.05-14**, acceptable because the applicant revised and provided markups in DCA Part 2, Tier 2, Revision 2, Section 3.8.5.4 and Table 1.8-2 to add COL Item 3.8-3 calling 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. The applicant's response also met DSRS Acceptance Criterion 3.8.5.II.4. Therefore, **RAI 8963, Question 03.08.05-14**, is resolved and closed.

3.8.5.4.3.5 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 conducted a Phase 2 regulatory audit from December 3rd to 7th, 2018 (ADAMS Accession No. ML18324A551), to review the applicant's related documents and to obtain additional information of the NuScale design of the seismic Category I structures. During the audit, the staff performed spot reviews of the following applicant 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 degree and 0.00784 degree 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 is expected to address this issue in the response to **RAI 8963, Question 03.08.05-6**, which the applicant submitted in a letter dated December 20, 2018 (ADAMS Accession No. ML18354B330), for the staff's review and confirmation.

- 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, and as discussed below with regard to **RAI 8963, Question 03.08.05-22,** dated August 23, 2018 (ADAMS Accession No. ML18235A280), 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. In addition, DCA Part 2, Tier 2, Revision 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 75 ksf (3.59 MPa), as specified in DCA Part 2, Tier 2, Revision 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 are acceptable because it is a realistic engineering practice to consider adjacent finite elements’ demand forces and moments when 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, to **RAI 8963, Question 03.08.05-13**, (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 to **RAI 8963, Question 03.08.05-12,** (ADAMS Accession No. ML18180A404), below, and provided markups to DCA Part 2, Tier 2, Revision 2, Section 3.7.2.1.1.1, “SAP2000,” and Section 3.7.2.1.2.5, “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 met 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 found that the applicant removed the listed COF values used in the stability assessments of the RXB and CRB from Revision 2 of DCA Part 2, Tier 2, Table 2.0-1. Therefore, the staff requested for a tabulation of the COF values in DCA Part 2, Tier 2, Revision 3, Table 2.0-1. The staff also requested the applicant to correctly reflect the COFs used in the calculations in the DCA. The applicant agreed to incorporate both requests in Revision 3 of DCA Part 2, Tier 2.

- 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. However, DCA Part 2, Tier 2, does not provide any information related to the requirement of this construction load of concrete pour pressure of 600 psf (28.72 kPa) based on ACI 347. Therefore, the staff requested the applicant to describe this loading condition in DCA Part 2, Tier 2, and the applicant agreed to do so. The staff also resorted to this calculation to address the applicant's response to **RAI 8963, Question 03.08.05-23**, below.

- The staff reviewed the calculation EC-F010-1731, Revision 3, "Structural Analysis for NuScale Building Using SAP2000," to determine how the applicant considers the cracked concrete stiffness properties in the models to account for any concrete cracking during a seismic event. In this report, the applicant describes how to reduce the stiffness components to perform the seismic analyses under cracked concrete condition. The applicant provides tables tabulating cracked stiffness rigidity equation of "axial (E_cA_g), shear (G_cA_w) and flexural (E_cI_g)." The applicant considers 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 approach that reduced the concrete element thickness in its seismic analyses. However, the applicant describes the level of concrete cracking considered for the cracked-concrete properties are based on the guidance from Table 3-1 in ASCE 43-05 Section 3.4.1, in Revision 2 of DCA Part 2 Tier 2, Sections 3.8.4.4.1 "Reactor Building Analysis," and 3.8.4.4.2 "Control Building Analysis." The applicant also provides Table 3.7.1-7 "Effective Stiffness Reinforced Concrete Members," in DCA Part 2 Tier 2, Section 3.7.1.2.2 "Structural Damping," which is as same as Table 3-1 in ASCE 43-05 Section 3.4.1. Therefore, it appears that there is a discrepancy in the use of concrete cracked-properties, what the applicant describes in the DCA Part 2 and Tier 2, and what the applicant used in the calculations. The staff identified this issue subsequently during the review of the audit notes/records. Thus, this issue was not identified and discussed during the audit-phase. During the public teleconference on January 23, 2019, the staff requested the applicant to describe the application of the concrete cracked-properties in analyses. The applicant submitted a response in a letter dated February 14, 2019 (ADAMS Accession No. ML19045A493) addressing this issue by providing a new Table 3.7.1-7a "Effective Stiffness Changes of Cracked Reinforced Concrete Finite Element Model Members," and referring this table in Section 3.7.1.2.2 "Structural Damping in DCA Part 2, Tier 2, Revision 3. Based on the review, the staff found the applicant response acceptable because the applicant clearly identified the crack concrete properties used in the finite element models in Table 3.7.1-7a and referenced that table in Section 3.7.1.2.2 and provided associated markups in its response letter referenced above. The applicant's response also met DSRS Acceptance Criterion 3.7.1.II.4. The staff is tracking this issue as **Confirmatory Item 03.08.05-1.**

3.8.5.4.3.5.1 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.

3.8.5.4.3.5.2 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 combine 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. 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." Contrary to that statement, DCA Part 2, Tier 2, page 3.8-59, states, "Figure 3.8.5-1 shows the SAP2000 model. The area elements shown in light red tinge are shell elements representing the base slab." Based on the above statements, it was not clear to the staff what types of elements were used to create the RXB basemat models in SAP2000. Similarly, the staff was unclear as to what types of elements were used in the SASSI2010 model to calculate forces and moments in the basemat. Therefore, the staff issued **RAI 8963, Question 03.08.05-11**, asking the applicant to describe all the elements used in the SAP2000 and SASSI2010 RXB basemat models, including the boundary elements between the basemat and soils.

The staff reviewed the applicant's response, dated October 17, 2017 (ADAMS Accession No. ML17290B267), to **RAI 8963, Question 03.08.05-**11. The applicant also submitted a supplemental response dated February 21, 2018 (ADAMS Accession No. ML18052B566), to **RAI 8963, Question 03.08.05-11**, to address the discussions during the January 30, 2018, public meeting.

In its responses, the applicant explained that the RXB basemat in the SAP2000 models (standalone and triple building) are composed of two layers of 5-foot (1.524-meter) thick solid elements with fixed-based boundary conditions to determine static bearing pressures and settlements. The applicant also stated 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 5-foot (1.524-meter)-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. DCA Part 2, Tier 2, Revision 2, Figures 3.8.5-1–3.8.5-7, show static and seismic RXB base pressures contours and static and seismic RXB bending moments.

Based on its review, the staff finds the response to **RAI 8963, Question 03.08.05-11**, acceptable because the applicant stated that two layers of 5-foot (1.524-meter)-thick concrete solid elements were used 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 address static and seismic load combinations to determine static and seismic bearing pressures. The applicant also stated that two 5-foot (1.524-meter)-thick layers of concrete solid elements were replaced by a single layer of shell element in the standalone RXB model using the SAP2000 software, 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 provided markups consistent with this response addressing the RXB model in DCA Part 2, Tier 2, Revision 2, Section 3.8.5.4.1.2. Furthermore, the applicant's response to **RAI 8963, Question 03.08.05-12**, below, with markups, clearly describes and provides additional information related to the models used in the design of the RXB and CRB basemats. Therefore, **RAI 8963, Question 03.08.05-11**, is resolved and closed.

DCA Part 2, Tier 2, Section 3.8.5.4.1.2, "RXB Basemat Analysis Model Description," states the following:

> *The static forces and moments in the basemat are calculated with both the standalone and the combined building SAP2000 models.*
>
> *…The seismic forces, moments and stresses in all structural elements such as walls, pilasters, and basemat were calculated using the standalone and combined SASSI2010 models. The enveloped base pressures were applied to the solid foundation model to evaluate the responses. To be consistent with the SASSI2010 analysis, absolute values of all responses obtained by applying base pressures from SASSI2010 were used together with the fixed end forces and moments from walls and pilasters to arrive at the seismic demands.*

It was not clear to the staff which structures or portions of the structures are included in the standalone and the combined building SAP2000 and SASSI2010 models, and whether there is a standalone basemat model. Therefore, the staff issued **RAI 8963, Question 03.08.05-12**, asking the applicant to clarify which structures or portions of the structures are included in the standalone and the combined building SAP2000 and SASSI2010 models, and to clarify whether there is a standalone basemat model.

In its response dated October 17, 2017 (ADAMS Accession No. ML17290B267), to **RAI 8963, Question 03.08.05-12**, the applicant provided two tabulations of a building model and basemat model used to determine the internal forces and moments of RXB basemat using the SAP2000 and SASSI2010 software.

In its response, the applicant further 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 10 feet (3.05 meters) above the basemat are deleted, (3) walls, pilasters, and columns are restrained in all six directions at 10 feet (3.05 meters) above the basemat, and (4) the envelope soil bearing pressures, because of static and dynamic loads, are applied as uniform pressures to the basemat. However, the staff determined that DCA Part 2, Tier 2, did not describe this new partial RXB model.

In a public meeting on January 10, 2018, the staff requested the applicant to include the information in its response to **RAI 8963, Question 03.08.05-12**, to clearly address the modeling and associated software used in the design of the RXB basemat. In its supplemental response dated June 29, 2018 (ADAMS Accession No. ML18180A404), to **RAI 8963, Question 03.08.05-12**, the applicant provided the markups in DCA Part 2, Tier 2, Revision 2, Sections 3.7.2.1.1.1 and 3.7.2.1.2.5, stating that the RXB and CRB basemats are designed using combinations of different models 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. The applicant provided new tables summarizing the RXB and CRB building models and basemat models used for the basemat designs. In its response, the applicant also provided markups in DCA Part 2, Tier 2, Revision 2, to include new 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," that clearly describe in tabular format the software used (SASSI2010, SAP2000), buildings included in the model, number of layers and types of element used in basemat model, and results used for the designs of RXB and CRB basemats.

Based on its review, the staff finds the response to **RAI 8963, Question 03.08.05-12**, acceptable because the applicant described how the RXB and CRB basemats are designed and provided associated tables summarizing the building and basemat models used in the design of the basemats. In its response, the applicant also provided the markups in DCA Part 2, Tier 2, Revision 2, Sections 3.7.2.1.1.1 and 3.7.2.1.2.5 and associated tables (listed above), tabulating software (SASSI2010, SAP2000), buildings included in the model, number of layers and types of element used in basemat model, and results used for the designs of the RXB and CRB basemats. Therefore, **RAI 8963, Question 03.08.05-12**, is resolved and closed.

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 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 ANYSY 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 node-j when the gap is closed and consequently transmitting compression only. The respective acceleration time histories from the SASSI2010, for Soil Types 7, 8, and 11, were applied uniformly to all the boundary nodes in the ANSYS model. 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, Revision 2, and confirmatory review performed during the audit (ADAMS Accession No. ML18324A551), the staff concludes that the applicant provided sufficient information to describe the RXB SAP2000, SASSI2010, and ANSYS models.

3.8.5.4.3.5.3 Analysis of Control Building Basemat

DCA Part 2, Tier 2, Revision 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 ends 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.

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. 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 met DSRS Acceptance Criterion 3.8.5.II.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 is 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, "RXB Stability Evaluation Input Parameters," gives the COF between walls and soil and between basemat bottom surface and soil. However, the applicant did not tabulate these COF values in DCA Part 2, Tier 2, Table 3.8.5-9, "CRB Stability Evaluation Input Parameters." Therefore, the staff issued **RAI 8963, Question 03.08.05-8**, asking the applicant to provide the CRB COF values between walls and soil and between basemat bottom surface and soil in DCA Part 2, Tier 2, Table 3.8.5-9.

In its responses dated October 17, 2017 and February 21, 2018 (ADAMS Accession Nos. ML17290B267 and ML18052B566, respectively), to **RAI 8963, Question 03.08.05-8**, the applicant revised DCA Part 2, Tier 2, Table 3.8.5-9, tabulating values of COF between wall and soil and between basemat and soil (static analysis and dynamic analysis). The applicant also provided editorial markups correcting the inconsistent values of COF throughout DCA Part 2, Tier 2. Furthermore, the staff conducted a Phase 2 regulatory audit December 3rd to 7th, 2018 (ADAMS Accession No. ML18324A551) to review the applicant's documents, including DCA Part 2, Tier 2, Revision 2, Chapters 1 and 2, to obtain additional information to the responses to **RAI 8963, Question 03.08.05-8**. During the audit, the staff concluded, and the applicant agreed to revise the values of COF "between soil and basemats," and "between soil and walls," of the RXB and CRB in DCA Part 2, Tier 2, Table 2.0-1, and to provide appropriate markups in DCA Part 2, Tier 2, Revision 3, Section 3.8.5.

During the audit, the staff reviewed the documents and had discussions with the applicant related the values of COF used in the stability analyses in DCA Part 2, Tier 2. To resolve the editorial issue in **RAI 8963, Question 03.08.05-8**, the applicant agreed to tabulate the values of COF "between soil and basemats" and "between soil and walls" of the RXB and CRB in DCA Part 2, Tier 2, Table 2.0-1, and to provide appropriate markups in DCA Part 2, Tier 2, Revision 3, Section 3.8.5. The staff is tracking **RAI 8963, Question 03.08.05-8**, as **Confirmatory Item 03.08.05-2**.

3.8.5.4.3.5.4 Control Building Basemat Nonlinear Analysis Model Description

DCA Part 2, Tier 2, Revision 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 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, "Control Building Sliding and Uplift Displacements for Soil Type 7 and 11." 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 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 29.399 psi (0.203 MPa) 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 provides 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 3.85-10, “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 from the response to **RAI 8963, Question 03.08.05-14**, above.

Based on the information provided in DCA Part 2, Tier 2, Revision 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 met 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 groundwater or floodwater 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 met DSRS Acceptance Criterion 3.8.5.II.3.

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 178,467 kips (80,951 mt) and 3,059 kips (1,387 mt), 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.

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 considers 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 ($M_{N\text{-}S} = M_1+M_2+M_3$ and $M_{E\text{-}W} = M_4+M_5+M_6$) 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.F.

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 found 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 diving 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, “SAP2000 Model for Settlement,” 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 the stiffness of soil further reduced 50 percent to amplify the effect of differential movements or settlements. The applicant 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.*

It was not clear to the staff whether the reduced soil stiffness values are extended to the size of the triple building (RWB+RXB+CRB) basemats or extended for the entire soil model shown in

DCA Part 2, Tier 2, Figure 3.8.5-41, “SAP2000 Model for Settlement.” Therefore, the staff issued **RAI 8963, Question 03.08.05-13**, asking the applicant to describe whether the reduced soil stiffness values are extended to the size of the triple building (RWB+RXB+CRB) basemats or extended for the entire soil model shown in DCA Part 2, Tier 2, Figure 3.8.5-41.

The applicant submitted its original response to **RAI 8963, Question 03.08.05-13**, on October 17, 2017 (ADAMS Accession No. ML17290B267). In response to the discussions at the February 14, 2018, public meeting, the applicant submitted its first supplemental response on April 4, 2018 (ADAMS Accession No. ML18094B106). Finally, in response to the discussions at the May 29, 2018, and July 10, 2018, public meetings, the applicant submitted its second supplemental response on August 16, 2018 (ADAMS Accession No. ML18228A859), describing that it 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, 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 response, the applicant also described and provided moment and shear capacities of RXB (10-foot, 0-inch-thick basemat) and CRB (5-foot, 0-inch-thick basemat) foundations. The applicant referred to DCA Part 2, Tier 2, Revision 2, Appendix 3B, Table 3B-36, which gives the moment and shear capacities of a 5-foot (1.524-meter)-thick CRB basemat, and DCA Part 2, Tier 2, Revision 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 conducted a Phase 2 regulatory audit from December 3rd to 7th, 2018 (ADAMS Accession No. ML18324A551), to review the applicant’s documents and to obtain additional information on the responses to **RAI 8963, Question 03.08.05-13**. During the audit, the staff reviewed calculation ER-F010-4135, Revision 3, and determined that the applicant’s conservatively performed calculation considers 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 foundations settlements. In its response, the applicant also provided the associated markups in DCA Part 2, Tier 2, Revision 2, Section 3.8.5.5.5, describing the extend 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 responses and calculation, the staff finds the applicant’s response to **RAI 8963, Question 03.08.05-13**, acceptable because during the audit, the staff reviewed the associated calculation and 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 incorporated the markups in DCA Part 2, Tier 2, Revision 2, Section 3.8.5.5.5 (ADAMS Accession No. ML18310A254). The applicant’s responses also met DSRS Acceptance Criterion 3.8.5.II.4. Therefore, **RAI 8963, Question 03.08.05-13**, is resolved and closed.

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 factors of safety 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 a 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 15 feet (4.672 meters) 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 sliding was insignificant. The applicant also performed nonlinear sliding, overturning, and uplift analyses for the CRB to show sliding, overturning, and uplift are insignificant.

The applicant stated that the bearing capacity of the soil should provide a FOS of 3.0 for the static bearing pressure and a FOS of 2.0 for dynamic bearing pressure. In addition, the maximum allowable differential settlement for RXB and CRB is 1 inch (2.54 centimeters) total, or 1/2 inch (12.7 millimeters) per 50 feet (15.24 meters) in any direction at any point in either structure.

DCA Part 2, Tier 1, Table 5.0-1, provides an acceptance criteria for a tilt settlement of the basemats as 1/2 inch (12.7 millimeters) per 50 feet (15.24 meters) in any direction. However, DCA Part 2, Tier 2, Section 3.8.5.6.1, provides a tilt settlement of 1/2 inch (12.7 millimeters) per 50 feet (15.24 meters) in any direction for the RXB and CRB basemats. The tilt settlement values listed in DCA Part 2, Tier 1 and Tier 2, are inconsistent. Therefore, on August 12, 2017, the staff issued **RAI 8964, Question 03.08.05-1** (ADAMS Accession No. ML17224A004), asking the applicant to provide consistent tilt settlement values throughout DCA Part 2, Tier 2.

In its response dated October 11, 2017 (ADAMS Accession No. ML17284A859), to **RAI 8964, Question 03.08.05-1**, the applicant confirmed that the maximum allowable tilt settlement for the RWB, RXB, and CRB shall be 1 inch (2.54 centimeters) total or 1/2 inch (1.27 centimeters) per 50 feet (15.24 meters) 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 1/2 inch (1.27 centimeters) between the RXB and CRB, and the RXB and RWB. The applicant also provided markups that consistently tabulate the values of the maximum allowable tilt settlement and differential settlement in DCA Part 2, Tier 1, Revision 1, Table 5.0-1, and DCA Part 2, Tier 2, Table 2.0-1, for the RXB, CRB, and RWB basemats.

Based on its review, the staff finds the response to **RAI 8964, Question 03.08.05-1**, acceptable because the applicant tabulated the values of maximum allowable tilt settlement and differential settlement values consistently in DCA Part 2, Tier 1, Revision 1, Table 5.0-1, and DCA Part 2, Tier 2, Table 2.0-1, for the RXB, CRB, and RWB, and the markups are consistent with the response in DCA Part 2, Tier 1, Revision 1, and DCA Part 2, Tier 2. Therefore, **RAI 8964, Question 03.08.05-1**, is resolved and closed.

DCA Part 2, Tier 2, Section 3.8.5.6.1 states, “reinforcing pattern described above.” However, the staff could not find any description of reinforcement in the previous text. Therefore, the staff issued **RAI 8963, Question 03.08.05-15**, asking the applicant to address reinforcing pattern as referenced.

In its response dated October 17, 2017 (ADAMS Accession No. ML17290B267), to **RAI 8963, Question 03.08.05-15**, the applicant provided markups in DCA Part 2, Tier 2, Revision 1, Section 3.8.5.6.1, by deleting the whole paragraph that included, “reinforcing pattern described above.” During the January 30, 2018, public meeting, the applicant submitted a supplemental response dated February 21, 2017 (ADAMS Accession No. ML18052B566), to **RAI 8963, Question 03.08.05-15**, providing markups with additional removals in DCA Part 2, Tier 2, Revision 1, Section 3.8.5.6.1. During its review of DCA Part 2, Tier 1, Section 3.8.5.6.1, to respond to **RAI 8963, Question 03.08.05-15**, the applicant identified additional text that related to the design aspects of the RXB, rather than to its stability assessment. Therefore, the applicant removed the text in DCA Part 2, Tier 2, Revision 1, Section 3.8.5.6.1, even though this change was not pertinent to the original RAI.

Based on its review, the staff finds the response to **RAI 8963, Question 03.08.05-15**, acceptable because the applicant deleted all the unrelated paragraphs in DCA Part 2, Tier 2, Revision 1, Section 3.8.5.6.1 (ADAMS Accession No. ML18086A046). Therefore, **RAI 8963, Question 03.08.05-15**, is resolved and closed.

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 factor of safety 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.

The applicant stated, “As shown in Section 3.8.5.4.1.4, … The FOS for flotation is shown in Table 3.8.5-5 for each of the 16 cases considered, … For each of the cases, an acceptable FOS for overturning was met.” The staff finds that the applicant incorrectly cited the reference to DCA Part 2, Tier 2, Section 3.8.5.4.1.4, because Section 3.8.5.4.1.4 is entitled “Control Building Basemat Nonlinear Analysis Model Description,” and the correct reference should be to DCA Part 2, Tier 2, Section 3.8.5.5.1, “Flotation and Uplift Stability Analysis Approach.” Therefore, the staff issued **RAI 8992, Question 03.08.05-3**, asking the applicant to address this inconsistency.

In its response dated October 11, 2017 (ADAMS Accession No. ML17284A810), to **RAI 8992, Question 03.08.05-3**, the applicant provided markups in DCA Part 2, Tier 2, Revision 1, Section 3.8.5.6.1.1, “RXB Uplift,” to correctly cite Section 3.8.5.5.1, instead of Section 3.8.5.4.1.4.

Based on its review, the staff finds the response to **RAI 8992, Question 03.08.05-3**, acceptable because the applicant correctly cited Section 3.8.5.5.1 in DCA Part 2, Tier 2, Revision 1, Section 3.8.5.6.1.1, and the markup is consistent with the response in DCA Part 2, Tier 2 Revision 1 (ADAMS Accession No. ML18086A046). Therefore, **RAI 8992, Question 03.08.05-3**, is resolved and closed.

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

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 the seismic reactions are much less than the total dead weight reaction of 471,487 kips (213,863 mt). Therefore, the applicant concluded that the net reactions are always in compression.

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 met DSRS Acceptance Criterion 3.7.2.II.4 of “assuring the ground contact ratio is equal to or greater than 80 percent.”

The applicant referred to vertical seismic reaction time histories used in the dynamic uplift evaluation. However, the applicant did not discuss whether the horizontal components of the input ground motions would impact the RXB uplift ratio. Therefore, the staff issued **RAI 8963, Question 03.08.05-16**, asking the applicant to clarify whether the horizontal components of the input ground motions were used in the calculation of RXB uplift ratio (if any).

In its response to **RAI 8963, Question 03.08.05-16**, dated October 17, 2017 (ADAMS Accession No. ML17290B267), the applicant revised 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,” to describe three acceleration components, X (EW), Y (NS), and Z (vertical), for each of the CSDRS and CSDRS-HF compatible seismic inputs used in the calculations of RXB and CRB uplift ratios.

Based on review, the staff finds the response to **RAI 8963, Question 03.08.05-16**, acceptable because the applicant described in the calculations of RXB and CRB uplift ratios, three acceleration components, X (EW), Y (NS), and Z (vertical), for each of the CSDRS and CSDRS-HF seismic inputs used, and the markups are consistent with the response in DCA Part 2, Tier 2, Revision 1, Sections 3.8.5.6.1.1.1 and 3.8.5.6.2.1.1 (ADAMS Accession No. ML18086A046). Therefore, the staff is tracking **RAI 8963, Question 03.08.05-16**, as **Confirmatory Item 03.08.05-3**.

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 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).

DCA Part 2, Tier 2, Table 3.8.5-11 “Reactor Building Sliding Displacements for Soil Type 7, 8 and 11 (Dead Weight + Buoyancy),” summarizes the sliding displacement results, which indicate that the deeply embedded RXB experiences less than 1/8 inch (0.318 centimeters) of sliding horizontal displacement.

DCA Part 2, Tier 2, Section 3.8.5.6.1.2, states “…a nonlinear sliding analysis has been performed to show that sliding is insignificant.” However, the applicant did not provide reasons to justify that the horizontal sliding results are insignificant. Therefore, the staff issued **RAI 8963, Question 03.08.05-18**, asking the applicant to describe the nonlinear analysis method and justify that the result of horizontal sliding is insignificant.

In its responses dated October 17, 2017 and February 21, 2018 (ADAMS Accession Nos. ML17290A267 and ML18052B566), to **RAI 8963, Question 03.08.05-18**, 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, and tabulated the sliding results in DCA Part 2, Tier 2, Table 3.8.5-12, where the maximum sliding displacements are 0.11 inch (0.28 centimeters). The applicant concluded that the displacements at these magnitudes would not cause any structural damages to RXB and CRB structures.

Based on its review, the staff finds the applicant’s response 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 applicant also tabulated the results in DCA Part 2, Tier 2,

Table 3.8.5-12 and Table 3.8.5-13, “Control Building Sliding and Uplift Displacements for Soil Type 7 and 11,” where the staff agreed and concluded that the results did not show any structural damage to RXB and CRB structures. Furthermore, the staff conducted a Phase 2 regulatory audit from December 3rd to 7th, 2018 (ADAMS Accession No. ML18324A551), to review the applicant’s documents and to obtain additional information to the response to **RAI 8963, Question 03.08.05-18**. During the audit, the staff 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, met 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 the applicant’s responses to **RAI 8963, Question 03.08.05-18**, acceptable, and the question is resolved and closed.

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 factor of safety 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 met 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 conducted a Phase 2 regulatory audit from December 3rd to 7th, 2018 (ADAMS Accession No. ML18324A551), to review the applicant’s documents and to obtain additional information. During the audit, the staff reviewed calculation ER-F170-3932, Revision 2. 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 stability evaluations are small. The staff concluded the stability displacements of deeply embedded structures would be a minor consideration under the seismic conditions and, therefore, met 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. 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 1/64 inch (0.4 centimeter) and concluded that the potential uplift is insignificant.

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

Dynamic Control Building Uplift Ratio

DCA Part 2, Tier 2, Section 3.8.5.6.2.1.1, “Dynamic CRB Uplift Ratio,” 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 3.8.5-15, “Seismic Vertical CRB Base Reactions and Dead Weight,” gives the results of seismic vertical and dead weight and shows the seismic reactions are much less than the total dead weight reaction of 45,680 kips (22,080 mt). 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, 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, met 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 0.006 inch (0.01524 centimeter) at location A, depicted in DCA Part 2, Tier 2, Figure 3.8.5-48. DCA Part 2, Tier 2, Revision 2, Table 3.8.5-12, “Control Building Sliding and Uplift Displacements for Soil Type 7 and 11,” summarizes the sliding and uplifting results, in which the magnitudes of these displacements are insignificant.

Based on its review, the staff found that the applicant’s results are acceptable to show insignificant sliding displacements as the CRB is a deeply embedded structure and; therefore, is consistent 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 3.8.5-13, shows that the deeply embedded CRB experiences less than 1/10 inch (0.254 centimeter) of overturning horizontal displacement and less than 1/64 inch (0.04 centimeter) 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 found 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 and; therefore is consistent 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 the bearing capacity of the soil should provide a FOS of 3.0 for the static bearing pressure and a 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 found 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 587,147 kips (266,325 mt), and the calculated footprint is 58,175 ft^2 (5,404.6 square meters (m^2)). This results in an average pressure of 10.1 ksf (0.484 MPa) and a FOS of 6.9 to the minimum soil bearing capacity of 75 ksf (3.59 MPa), specified in DCA Part 2, Tier 2, Table 2.0-1.

The dead weight of the CRB (based on static vertical gravity reaction (1 G_Z) and soil weight) is 75,779 kips (34,373 mt), with a base area of 11,800 ft^2 (1,096.3 m^2). This results in a static bearing pressure of 6.42 ksf (0.307 MPa). This value for the CRB static bearing pressure provides a FOS of 10.9 to the minimum soil bearing pressure of 75 ksf (3.59 MPa) shown in DCA Part 2, Tier 2, Table 2.0-1.

DCA Part 2, Tier 2, Section 3.8.5.6.3, provides static bearing pressures of 10.1 ksf (0.484 MPa) and 6.42 ksf (0.307 MPa) and dynamic bearing pressures of 4.6 ksf (0.22 MPa) and 5.32 ksf (0.255 MPa) for the RXB and CRB basemats, respectively. DCA Part 2, Tier 2, Figure 3.8.5-3, "Seismic Base Pressure Contours from SASSI2010 Analysis," shows the seismic bearing pressure contours from SASSI2010. The applicant determined that the highest bearing pressures are along the E-W edges of the RXB basemat and under the NPMs. However, it is not clear to the staff how the applicant determined 4.6 ksf (0.22 MPa) from DCA Part 2, Tier 2, Figure 3.8.5-3. Therefore, the staff issued **RAI 8963, Question 03.08.05-22**, asking the applicant to describe the reason(s) why the dynamic bearing pressures of the CRB are larger than the dynamic bearing pressures of the RXB, and to provide figures of static and seismic basemat pressure contours for the CRB and CRB tunnel.

The applicant submitted its original response on October 17, 2017 (ADAMS Accession No. ML172290B267), to **RAI 8963, Question 03.08.05-**22. In response to the discussions at the February 14, 2018, public meeting, the applicant submitted its first supplemental response on April 4, 2018 (ADAMS Accession No. ML18094B106). Finally, in response to the discussions at the June 12, 2018, public meeting, the applicant submitted its the second supplemental response on August 23, 2018 (ADAMS Accession No. ML18235A280), describing a new calculated result of 2.3 ksf (0.11 MPa) for the average dynamic bearing pressures of the entire CRB basemat and providing markups of DCA Part 2, Tier 2, Revision 2, Section 3.8.5.6.3.

The applicant responded 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 also described and provided Figures 3.8.5-2a and 3.8.5-3, showing the CRB basemat contour pressures for static and seismic loads. The applicant elaborated on the CRB tunnel basemat assessments of the SAP2000 analyses for obtaining the moments for design and settlement values for the CRB tunnel in DCA Part 2, Tier 2, Section 3.8.5.6.4, "Settlement," and added Table 3.8.5-17, "CRB Tunnel Foundation Corner Displacements," and Table 3.8.5-18, "CRB Tunnel Differential Settlement over 50 Feet and Tilt Angle," in DCA Part 2, Tier 2, Revision 2. Furthermore, the staff conducted a Phase 2 regulatory audit from December 3rd to 7th, 2018 (ADAMS Accession No. ML18324A551), to review the applicant's documents and obtain additional information in response to **RAI 8963, Question 03.08.05-22**. During the audit, the staff reviewed calculation ER-F170-3858, Revision 2, "Seismic Soil-Structure Interaction Analysis of NuScale CRB for Structural Responses," and 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, Revision 2, Table 3.8.5-15, "Average Soil Bearing Pressures (Toe Pressures) along Edges of CRB Basemat," which are less than the minimum soil-bearing pressure capacity of 75 ksf (3.59 MPa), as specified in DCA Part 2, Tier 2, Revision 2, Table 2.0-1 and Section 2.5.4, "Stability of Subsurface Materials and Foundations."

Based on its review, the staff finds the applicant's response to **RAI 8963, Question 03.08.05-22**, acceptable because during the audit, the staff reviewed the associated calculation and confirmed that the applicant performed calculations to determine average dynamic bearing pressures of 2.3 ksf (0.11 MPa) for the 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. The applicant's response includes markups in DCA Part 2, Tier 2, Revision 2, Sections 3.8.5.4.1.3, 3.8.5.6.3, and 3.8.5.6.4; Tables 3.8.5-18 and 3.8.5-19; and Figures 3.8.5-2a, 3.8.5-4a, 3.8.5-5a, 3.8.5-6a, and 3.8.5-7a. The applicant's response also met DSRS Acceptance Criterion 3.8.5.II.4. Therefore, **RAI 8963, Question 03.08.05-22**, is resolved and closed.

3.8.5.4.5.4 Settlement

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. 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 1-3/4 inches (4.45 centimeters) on the west end and approximately 2 inches (5.08 centimeters) on the east end, and the differential settlement of 0.25 inch (0.64 centimeter) is less than 1 inch (2.54 centimeters), 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 1-3/4 inches (4.445 centimeters) on the west end and approximately 1 inch (2.54 centimeters) on the east end, and the differential settlement of 0.75 inch (1.91 centimeters) is less than 1 inch (2.54 centimeters), 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 1/4 inch (0.64 centimeter). DCA Part 2, Tier 2, Revision 2, Table 3.8.5-17, “CRB Tunnel Foundation Corner Displacements,” 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 2.0 inches (5.08 centimeters). DCA Part 2, Tier 2, Revision 2, Table 3.8.5-18, “CRB Tunnel Differential Settlement over 50 Feet and Tilt Angle,” 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 50-foot (15.24-meter)-long differential settlement in the E-W direction is 0.36 inch (0.914 centimeter).

Based on its review, the staff found that the applicant’s tabulated results of settlements are acceptable in DCA Part 2, Tier 2, Revision 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, Revision 2, Tables 3.8.5-17 and 3.8.5-18 for the CRB tunnel foundation. The applicant submitted its response to **RAI 8963, Question 03.08.05-6**, on December 20, 2018 (ADAMS Accession No. ML18354B330). The applicant submitted supplemental response on March 28, 2019 (ADAMS Accession No. ML19087A330) for introducing headed reinforcing bars in Section 3.8.5.1 “Description of Foundations,” of DCA Part 2, Tier 2, Revision 3 that deviated from the reinforcing bars with standard hooks. Even though, the headed bars are not included in the provisions of ACI 349-06. In its response, the applicant describe that the headed reinforcing bars have equivalent or superior performance compare to the reinforcing bars with standard hooks and offers less congestion, which facilitates concrete consolidation and faster installation times, reducing placement costs. Moreover, the headed reinforcement bars were introduced in ACI 318 in the 2008 edition, followed by ACI 349-13. Based on the review, the staff found that the applicant supplemental response is acceptable since the applicant provided appropriate level of information describing the application of headed reinforcing bars in Section 3B1.1.3 “Wall and Slab Design Approach,” and referred “American Concrete Institute, "Building Code Requirements for Structural Concrete and Commentary," ACI 318-08, Farmington Hills, MI,” in Section 3B4 “References,” in DCA Part 2, Tier 2, Revision 3. Therefore, the staff considers **RAI 8963, Question 03.08.05-6**, acceptable since it is consistent with the DSRS 3.8.5.II.4.H and is being tracked as **Confirmatory Item 03.08.05-4**. The applicant shall incorporate information related to the application of headed reinforcing bars in DCA Part 2, Tier 2, Revision 3.

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 considers the thermal loads to be minor loads and self-relieving because of concrete cracking and moment distribution.

DCA Part 2, Tier 2, Section 3.8.5.6.5, states, “An explicit analysis considering these loads (Thermal Loads) has not been performed, as thermal loads are a minor consideration.” DCA Part 2, Tier 2, Section 3.8.4.4.1, states, “a fixed base model was created in ANSYS to evaluate the effects of thermal loads on the structure.” Therefore, the staff issued **RAI 8963, Question 03.08.05-5**, asking the applicant to address this discrepancy.

In its response dated October 17, 2017 (ADAMS Accession No. ML17290B267), to **RAI 8963, Question 03.08.05-5**, the applicant revised DCA Part 2, Tier 2, Section 3.8.4.4.1, by deleting the statement related to thermal loads on the RXB structure. The applicant considered thermal loads as minor loads and self-relieving because of concrete cracking and moment distribution.

Based on its review, the staff finds the applicant's response to **RAI 8963, Question 03.08.05-5**, acceptable because the applicant deleted the statement related to thermal loads on the RXB structure in DCA Part 2, Tier 2, Revision 1, Section 3.8.4.4.1 (, and the markup is consistent with this response. The staff also found with the applicant's description of thermal loads as minor loads and self-relieving is acceptable, and because of concrete cracking and moment distribution, and, thus, the description met DSRS Acceptance Criterion 3.8.5.II.4.B. Therefore, **RAI 8963, Question 03.08.05-5**, is resolved and closed.

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 found with the applicant's conclusion related to the main loads will be added after the completion of the RXB construction is acceptable since any loads induced by the construction sequence and differential settlements will be negligible in the design of RXB basemat. Similarly, loads induced by the construction sequence and differential settlements 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.

The staff conducted a Phase 2 regulatory audit from December 3rd to 7th, 2018 (ADAMS Accession No. ML18324A551), to review the applicant's related documents and obtain additional information on the design of the seismic Category I structures. During the audit, 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, DCA Part 2, Tier 2, does not provide any information related to the requirement of this construction load of concrete pour pressure of 600 psf (28.72 kPa) based on ACI 349. Therefore, the staff requested the applicant to describe this loading condition in DCA Part 2, Tier 2, and the applicant agreed to do so. The staff also looked to this calculation to address the applicant's response to **RAI 8963, Question 03.08.05-23**, below.

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 a FOS of 3.0 for the static bearing pressure and a FOS of 2.0 for the dynamic bearing pressure in DCA Part 2, Tier 2, Section 3.8.5.6.1.

DCA Part 2, Tier 2, Tables 3.8.5-14 and 3.8.5-16, provide the average toe pressures for RXB and CRB, respectively. The values shown in these tables are more than the dynamic bearing pressure capacity of 2.0, where the minimum soil bearing pressure capacity is 75 ksf (3.59 MPa), as specified in DCA Part 2, Tier 2, Table 2.0-1.

The staff finds that DCA Part 2, Tier 2, Tables 3.8.5-14 and 3.8.5-16, include the applicant's analyses and tabulation of the results of average toe pressures for the RXB and CRB, and show that they are all more than the dynamic bearing pressure capacity of 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 effect of settlements. Therefore, the staff issued **RAI 8963, Question 03.08.05-2**, requesting the applicant to provide information related to the settlements from soft-soil stiffness values along the edges of the seismic Category I structure basemats.

In its response to **RAI 8964, Question 03.08.05-2**, dated August 28, 2018 (ADAMS Accession No. ML18240A156), the applicant provided new Tables 3.8.5-7a and 3.8.5-7b and associated markups in DCA Part 2, Tier 2, Revision 2, Section 3.8.5.6.4, and referred to those tables, tabulating 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, Revision 2, Figure 3.8.5-10, "Edge and Center Nodes at Bottom of Foundations Selected for Building Settlement Assessment," also shows the locations of critical edge and center nodes at the bottom of the foundations. Furthermore, the staff conducted a Phase 2 regulatory audit from December 3rd to 7th, 2018 (ADAMS Accession No. ML18324A551), to review the applicant's documents and to obtain additional information in response to **RAI 8964, Question 03.08.05-2**. During the audit, the staff reviewed calculation ER-F010-4135, Revision 3, 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 response to **RAI 8964, Question 03.08.05-2**, acceptable because the applicant provided new DCA Part 2, Tier 2, Tables 3.8.5-7a and 3.8.5-7b, that give the 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 in DCA Part 2, Tier 2, Revision 2, Figure 3.8.5-10. The staff also confirmed the tabulated settlement values are below the maximum total value of 4 inches, as provided in DCA Part 2, Tier 2, Revision 2, Table 2.0-1. The applicant's response also met DSRS Acceptance Criterion 3.8.5.II.4. Therefore, **RAI 8964, Question 03.08.05-2**, is resolved and closed.

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 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. Therefore, the staff issued **RAI 8963, Question 03.08.05-23**, asking the applicant to describe the leak chase system in detail (e.g., safety categorization of the systems, location of the leak chase system collection room and collection tank, associated piping from the RXB basemat and design requirements).

In its response dated October 17, 2017 (ADAMS Accession No. ML17290B267), to **RAI 8963, Question 03.08.05-23**, 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, Section 3.2, classifies the pool leakage detection system, radioactive waste drain system, and liquid radioactive waste system as nonsafety-related and not risk-significant (B2).

The staff conducted a Phase 2 regulatory audit from June 14–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 conducted a second independent structural Phase 2 regulatory audit from December 3–7, 2018 (ADAMS Accession No. ML18324A551), to review applicant documents and to obtain additional information in response to **RAI 8963, Question 03.08.05-23**. During the audit, the staff 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. However, the applicant also stated 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. However, DCA Part 2, Tier 2, did not provide any information related to the requirement of this construction load of concrete pour pressure of 600 psf (28.72 kPa) based on ACI-347. Therefore, the staff requested the applicant to describe this loading condition in DCA Part 2, Tier 2, Revision 3, and the applicant agreed to do so.

Based on the review and audits performed, the staff determined the applicant's response to **RAI 8964, Question 03.08.05-23**, 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, including the identified concrete pour pressure of 600 psf (28.72 kPa) as the construction load in the load combinations. However, the staff requested the applicant to describe this loading condition in DCA Part 2, Tier 2, Revision 3, and the applicant agreed to do so. The staff considers **RAI 8964, Question 03.08.05-23**, acceptable and is tracking it as **Confirmatory Item 03.08.05-5**. The applicant shall incorporate concrete pour pressure of 600 psf (28.72 kPa) as the construction load in the load combinations in DCA Part 2, Tier 2, Revision 3.

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 5,000 psi (34.47 MPa). DCA Part 2, Tier 2, Section 3.8.4.6.1.1, also states that the concrete ingredients are cement, aggregates, admixtures, and water. 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, “Testing and Inservice Inspection Requirements,” describes the testing and inservice inspection 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 SER 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, “RXB ITAAC,” and Table 3.13-1, “CRB ITAAC.”

3.8.5.5 Combined License Information Items

Based on its response to **RAI 8963, Question 03.08.05-14**, the applicant added COL Item 3.8-3 in DCA Part 2, Tier 2, Revision 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

COL 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 are acceptable and meets the requirements described in Section 3.8.5.3 of this SER, except for the Confirmatory Items identified above that will be dispositioned based on the review and acceptance of the markups provided in Revision 3 of Section 3.8.5 in DCA Part 2, Tier 2, which will be provided by the applicant.

3.9 <u>Mechanical Systems and Components</u>

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 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 1 components and their supports, and experimental and inelastic analytical techniques.

The staff reviewed DCA Part 2, in accordance with SRP Section 3.9.1, "Special Topics for Mechanical Components," Revision 4, 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 the staff reviewed include the following:

- transients used in the design and fatigue analyses of all ASME BPV Code Class I 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)

<u>Testing Conditions</u>

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 for the Faulted Condition," identifies seismic Category I RCS items.

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

- 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 Pressure 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, “NuScale Power Module”; Section 2.2, “Chemical and Volume Control System”; Section 2.3, “Containment Evacuation System”; and Section 2.4, “Turbine Generator 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”; 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 cumulative usage factor calculation unless the OBE magnitude is smaller than one-third of the SSE, in accordance with 10 CFR Part 50, Appendix S. The OBE is defined to be one-third of the SSE and, therefore, is eliminated from explicit design or analysis, in accordance with 10 CFR Part 50, Appendix S. Therefore, the OBE is not used for primary stress evaluations and is not included in load combinations for the design of standard plant SSCs. However, the fatigue analysis of SSCs accounts for the effects of the OBEs.

The fatigue evaluation requires one SSE and five OBE events, which is equivalent to two SSEs with 10 maximum stress cycles each, for a total 20 full cycles. Alternatively, one SSE and five OBE events in the fatigue evaluation are also equivalent to 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, “Fatigue Evaluation of ASME BPV Code Class 1 Piping,” when derived in accordance with IEEE 344-2004, Annex D.1.

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 RCBP. 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, "ASME BPV Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports," 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. The staff issued **RAI 182-9039, Question 03.09.01-1**, requesting the applicant (1) to clarify whether the number of "events" in Table 3.9-1 is the number of occurrences for the named event and, if so, to change "events" to "Cycles," and (2) whether a note should be added to DCA Part 2, Tier 2, Table 3.9-1, to address the OBE for the fatigue analysis based on the discussion in DCA Part 2, Tier 2, Sections 3.7.3 and 3.12.4, with regard to 20 cycles of SSE or 312 cycles of one-third SSE (equivalent OBE) to be used in lieu of one SSE and five OBEs as required by the fatigue analysis.

In its response dated September 20, 2017 (ADAMS Accession No. ML17263B231), to **RAI 182-9039, Question 03.09.01-1**, the applicant failed to clarify the number of events from the event cycles that were discussed during the public teleconference on December 12, 2017. Therefore, the applicant provided supplemental information in a letter dated December 27, 2017 (ADAMS Accession No. ML17361A301), and included DCA Part 2, Tier 2, Table 3.9-1, marked with the proper description for events and occurrence cycles. The applicant also added an explanatory note to DCA Part 2, Tier 2, Table 3.9-1, for OBE fatigue cycles. The staff verified that the applicant incorporated this markup and explanatory note in DCA Part 2, Revision 1. Therefore, **RAI 182-9039, Question 03.09.01-1**, is resolved and closed.

GDC 1 requires 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 on the magnitude and frequency of temperature and pressure transients resulting from various operating conditions in the plant that may occur. The staff issued **RAI 182-9039, Question 03.09.01-2**, asking the applicant to provide the basis for assuming 90 cycles of turbine trip without bypass and 180 cycles of turbine trip with bypass and to confirm whether the pressure transients from the turbine trips cover the transients from the stop valve closure, which was considered to generate severe dynamic loads in PWR design.

In its response dated September 20, 2017 (ADAMS Accession No. ML17263B231), to **RAI 182-9039, Question 03.09.01-2**, the applicant stated that it selected the number of cycles for the turbine trip without bypass and turbine trip with bypass transients based on the predicted

NuScale probabilistic risk assessment initiating event frequencies, nuclear operating experience, and comparisons to recent DCAs. As DCA Part 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 of 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 the applicant’s response acceptable, and the staff concludes that **RAI 182-9039, Question 03.09.01-2**, is resolved and closed.

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 full power, at which the control systems are placed in automatic mode. DCA Part 2, Tier 2, Section 3.9.1.1.1, specifies that the maximum load ramp is limited to 0.5-percent full power per minute. It appears that one is the reverse of the other. These two are identified as Transient 3 and Transient 4, respectively. DCA Part 2, Tier 2, Table 3.9-1, describes the numbers of transient occurrences.

GDC 14 and 15 apply to SSCs important to safety, designed to 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 staff issued **RAI 182-9039, Question 03.09.01-3**, requesting the applicant to provide the basis as to why the ascent event has 700 occurrences and the descent event has 300 occurrences. The staff also requested that the applicant provide additional information and a justification for why the assumed numbers of occurrences for these two events were different than a previously certified standard PWR (i.e., AP1000), which used 500 cycles of occurrences for both events.

In its supplemental response dated March 19, 2018 (ADAMS Accession No. ML18078B295), to **RAI 182-9039, Question 03.09.01-3**, the applicant indicated that the previously accepted standard PWR conservatively used 600 occurrences for both events, while the last-approved application (AP1000) used 500 cycles for both events. The applicant also 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 discussed in the response to **RAI 9039, Question 03.09.01-3**. Although the selected number of power descent events is less than the 500 used by a previously approved design certification applicant, this difference is outweighed by the 200 additional power ascent events, which generate higher stresses in NPM components through faster temperature changes. This approach results in a more conservative impact on the affected NPM regions than assuming 500 power ascents and

500 power descents. Therefore, the staff concludes that the applicant's use of higher cycles for the higher-stress conditions is conservative for the fatigue design of Class 1 components and is acceptable. Therefore, **RAI 9039, Question 03.09.01-3**, is resolved and closed.

In accordance with GDC 14 and 15, SSCs important to safety are 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). DCA Part 2, Tier 2, Table 3.9-1, assumes 15 cycles for inadvertent PZR spray. The staff issued **RAI 182-9039, Question 03.09.01-4**, requesting the applicant to provide a justification for using 15 cycles instead of 30 cycles as is typical for a standard PWR plant for this transient.

In its response dated September 20, 2017 (ADAMS Accession No. ML17263B231), to **RAI 182-9039, Question 03.09.01-4**, the applicant stated that 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 chemical and volume control system design is an air-operated valve. 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 2.28×10^{-7} per hour, while the base line mean frequency in NUREG/CR-6928 for the air-operated valve failed to control is $3.0 \text{X} 10^{-6}$ per hour. Over a 60-year design life, this equates to 0.12 cycle (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 cycle per plant life. Therefore, the staff finds the applicant's response to **RAI 182-9039, Question 03.09.01-4**, to be acceptable because 15 cycles over 60 years is sufficiently bounding for the design analysis. Therefore, **RAI 182-9039, Question 03.09.01-4**, is resolved and closed.

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. The staff conducted a regulatory audit from March 20–April 27, 2018, of the computer codes in support of its reviews of DCA Part 2, Tier 2, Section 3.9.1, to ensure the computer programs used for the design of seismic Category I structures and components meet the requirements 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, EMDAQC, 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.*

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 all 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 was 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 issued **RAI 182-9039, Question 03.09.01-5**, requesting that the applicant identify the computer programs used for calculating stresses and cumulative usage factors for ASME BPV Code Class 1, 2, and 3 components, including staff-endorsed environmental effects on the fatigue methodology, and confirm that the analyses for ASME BPV Code, Section III, Class 1, components and piping for the fatigue evaluation include environmental effects in accordance with RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors,” and NUREG/CR-6909, “Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials, Final Report,” Revision 1, issued May 2018.

In its response dated September 20, 2017 (ADAMS Accession No. ML17263B231), to **RAI 182-9039, Question 03.09.01-5**, the applicant did not provide the computer code that is used to perform the fatigue analysis in accordance with RG 1.207, aligned with NUREG/CR-6909, to account for the environmental-assisted effects.

The staff conducted an audit for NuScale computer codes for ASME BPV Code Class 1, 2 and 3 components from March 20–April 27, 2018 (ADAMS Accession No. ML18074A079), to confirm that the computer programs used for 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 issue raised in **RAI 182-9039, Question 03.09.01-5**. 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. Therefore, the staff finds the applicant’s response acceptable, and **RAI 182-9039, Question 03.09.01-5**, is resolved and closed.

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

There are no COL information items for this area of review.

3.9.1.6 Conclusion

On the basis of the evaluations 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, Components, and Equipment

[SE issued separately ML19102A109]

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 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 includes 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. DCA Part 2, Tier 1, Tables 2.2-1 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, "ASME BPV Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," 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 NuScale ITAAC.

Design Specifications: The staff reviewed design specifications in the applicant's electronic reading room for the audit.

Technical Reports: The following NuScale TRs apply to DCA Part 2, Tier 2, Section 3.9.3, and the staff used information in these reports to make the safety findings:

- TR-0916-51502-P, "NuScale Power Module Seismic Analysis," Revision 1, dated September 28, 2018

- TR-0716-50439-P, "NuScale Comprehensive Vibration Assessment Program Technical Report," Revision 1, dated January 20, 2018

- TR-1016-51669, "NuScale Power Module Short-Term Transient Analysis," Revision 0, dated December 31, 2016

3.9.3.3 Regulatory Basis

SRP Section 3.9.3, "ASME BPV Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," Revision 3, issued April 2014, provides the relevant Commission regulations for this area of review and the associated acceptance criteria, ASME BPV Code summarized below, 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 it relates 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

- 10 CFR Part 50, Appendix A, 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 loss-of-coolant accidents

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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 anticipated operational occurrences

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, issued March 2007

- RG 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Supports," Revision 3, issued July 2013

- RG 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports," Revision 3, issued July 2013

- ASME Code, Section III, Subarticle NCA-3250, which requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems

3.9.3.4 Technical Evaluation

GDC 1, 2, and 4 and 10 CFR 50.55a require 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 LOCA. GDC 14 and 15 require that the RCBP 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 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. DCA Part 2, Tier 2, Table 3.9-2, "Pressure, Mechanical, and Thermal Loads," provides 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 3.9-4 relates 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 top support structure. 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 RVV and the reactor safety valves (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 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. 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 79-13, "Cracking in Feedwater System Piping," Revision 2, dated October 16, 1979; Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," dated June 22, 1988; and Bulletin 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, and transportation. On August 25, 2017, the staff issued **RAI 9021, Question 03.09.03-1** (ADAMS Accession No. ML17237C059), asking the applicant to clarify (1) whether DCA Part 2, should also discuss effects of loads from transportation (TR), load test (LT), hydrostatic and sloshing, CNV integrated leak rate test pressure, and hydrogen detonation, as listed in DCA Part 2, Tier 2, Table 3.9-2, but not discussed in DCA Part 2, Tier 2, Section 3.9.3.1.1, to ensure proper consideration of all applied loads, (2) whether Note 6 of DCA Part 2, Tier 2, Table 3.9-2, should also be included in DCA Part 2, Tier 2, Tables 3.9-3 through 3.9-14, because there is no feedwater pipe break (FWPB) and main steam pipe break (MSPB) inside the CNV, and (3) whether DCA Part 2, Tier 2, Section 3.9.3.1.1, should discuss loads from asymmetric cavity pressurization that result from HELBs in the primary coolant system and concurring with the blowdown loads, as described in TR-1016-51669. The staff developed the second question to specify that MSPB and FWPB outside the CNV are taken into account only for each load combination.

DCA Part 2, Tier 2, Revision 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. DCA Part 2, Tier 2, Revision 0, states that only Note 6 of Table 3.9-2 defines these pipe breaks. Therefore, the staff developed the question to clearly specify that MSPB and FWPB outside the CNV are taken into account only for each load combination. However, the applicant deleted Note 6 in DCA Part 2, Tier 2, Table 3.9-2, for Revision 2; as such, the second question is invalid.

In its response dated October 23, 2017 (ADAMS Accession No. ML17296B367), to **RAI 9021, Question 03.09.03-1**, the applicant stated that it revised DCA Part 2, Tier 2, Section 3.9.3.1.1, to discuss transportation (TR), load test (LT), hydrostatic and sloshing, asymmetric cavity pressurization, and hydrogen detonation loads. Appendix J, "Type A Integrated Leak Test," is not applicable to the NPM, in accordance with the justification provided in TR-1116-51962; therefore, the applicant removed this load from DCA Part 2, Tier 2, Table 3.9-2. The applicant also removed Note 6 from DCA Part 2, Tier 2, Table 3.9-2, and added it to the pipe break discussion in DCA Part 2, Tier 2, Section 3.9.3.1.1. DCA Part 2, Tier 2, Section 3.9.3.1.1, discusses where breaks occur, although this is not relevant to defining the load terms or how they are accounted for in the load combinations. The applicant revised DCA Part 2, Tier 2, Section 3.9.3.1.1, to discuss the location of postulated FWPB and MSPB. The cavity pressure discussed with regard to system operating transient and pipe break in DCA Part 2, Tier 2, Section 3.9.3.1.1, addresses the asymmetric cavity pressure. To provide clarity, the applicant added "asymmetric" before "cavity pressure." Additionally, the applicant added "occurring simultaneously with blowdown" to clarify that asymmetric cavity pressurization occurs concurrent with pipe break or spurious valve actuation. As a result, DCA Part 2, Tier 2, Section 3.9.3.1 and Table 3.9-2, have been revised as described in the response, and other changes are also shown in the markup provided with this response. The staff finds the applicant's changes clarify all applicable loads and load combinations to be considered and 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. The staff reviewed DCA Part 2, Tier 2, Revision 2, to ensure the applicant incorporated all markups in the response. As a result, **RAI 9021, Question 03.09.03-1**, is resolved and closed.

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 (see SER Section 3.7.5.3). 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 1, provides the results of the seismic analysis of the NPM. The staff reviewed the TR with respect to the seismic loads and stresses. As a result, the staff issued **RAI 9021,**

Question 03.09.03-2, asking the applicant to provide additional discussion and clarification for the following:

(1) Considering a subsystem seismic analysis for NPM as stated above, based on RG 1.61, Table 6, a 3-percent damping should be used in lieu of the 7-percent damping given in TR-0916-51502- P, for pressure vessel (i.e., CNV and RPV). Therefore, provide justification or test data for using higher than 3-percent damping for the CNV and RPV for the SSE event as given in Table 6 of RG 1.61.

(2) Provide the damping values used for the reactor poor water inside and outside the CNV and the basis for using these damping values in the SSE analysis.

(3) Confirm whether there is no uplift of the CNV and RPV during the SSE event, considering the buoyancy force and the bearing support at the bottom of CNV and RPV.

(4) Discuss whether and how the condition of the SSE event is not considered for the design of the CNV and RPV during the refueling outage.

(5) Describe boundary conditions of the 3-D NPM analysis where the prescribed acceleration time histories were applied.

In its response dated August 2, 2018 (ADAMS Accession No. ML18214A835) to **RAI 9021, Question 03.09.03-2(1)**, the applicant stated that the lowest specified damping value for the SSE event is 4 percent for welded steel or bolted steel with friction connections, rather than 3 percent. The composite damping value assigned to the NPM subsystem has been determined to use 4-percent damping, while RG 1.61, Revision 1, issued March 2007, employs 3 percent for SSE for the RPV. The applicant provided a further justification in a supplemental response dated October 16, 2018, (ADAMS Accession No. ML18289B091) to **RAI 9021, Question 03.09.03-2**, 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 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. Therefore, **RAI 9021, Question 03.09.03-2(1)**, is resolved and closed.

In its response to **RAI 9021, Question 03.09.03-2(2)**, 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. Therefore, the staff finds the applicant’s response acceptable, and **RAI 9021, Question 03.09.03-2(2)**, is resolved and closed.

In its response to **RAI 9021, Question 03-09-03-2(3)**, the applicant indicated that potential uplift of the NPM is captured through nonlinear contact with the rigid floor surface. 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 8-8, gives the results. The staff considered that the maximum uplift is too small to affect the structural integrity of NPM and, therefore, finds the applicant’s response acceptable.

In its response to **RAI 9021, Question 03.09.03-2(4)**, the applicant indicated that 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. NuScale updated TR-0916-51502 for 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. Therefore, **RAI 9021, Question 03.09.03-2(4)**, is resolved and closed.

In its response to **RAI 9021, Question 03.09.03-2(5)**, the applicant indicated 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 50 feet, 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. The applicant also stated 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 coincident, but separate, remote point, to allow for nonlinear contact of the CNV skirt and rigid floor. Acceleration boundary condition data generation is described in the TR Section 5.1. The staff finds the applicant's description is acceptable and confirmed that the NPM Seismic Analysis technical report TR-0916-51502 has been updated as described in the response. Therefore, **RAI 9021, Question 03.09.03-2(5)**, is resolved and closed.

3.9.3.4.2 Design and Installation of Pressure-Relief Devices

The RCS RSVs located on the RPV are designed as ASME Code, Section III, Class 1, pressure-relief, pilot-operated devices. There are two RSVs, which are not connected to any piping on their discharge sides and vent directly into the CNV. The RSV function is to prevent RCS pressure from exceeding 110 percent of design pressure under normal and abnormal conditions and to prevent the exceedance of service limits. The two valves, each with sufficient capacity to limit over pressurization of the RPV, are normally closed, are low leakage, and are used infrequently. The RCS and PZR steam space are sized to avoid an RSV lift for anticipated transients.

The ECCS valves are also located on the RPV and are part of the RCPB. These ECCS valves are seismic Category I components and designed as ASME BPV Code, Section III, Class 1, components. SER Section 6.3 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 issued **RAI 9021, Question 03.09.03-3**, requesting the applicant to (1) confirm whether and how these ASME BPV Code Class 1 components are qualified by analysis or test, or both, using static analysis or dynamic analysis, (2) discuss the loads considered for calculating fatigue cumulative usage factor and effects of the environment assisted fatigue for these valves, and (3) explain what damping values were used in the analysis.

In its response dated October 23, 2017 (ADAMS Accession No. ML17296B367), to **RAI 9021, Question 03.09.03-3(1)**, the applicant stated that for NuScale plants, ASME BPV Code Class 1 valves, such as RSVs and ECCS valves, are to be qualified to the requirements of IEEE 344-2004 and in compliance with ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants." In SER Section 3.9.6, the staff evaluates the acceptability of the qualification of ASME BPV Code Class 1 valves identified in **RAI 9021, Question 03.09.03-3(1)**.

In its response to **RAI 9021, Question 03.09.03-3(2)**, the applicant stated that 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 Code, Section III, Subsections NB-3200 or NG-3200, considering the effects of the LWR environment, in accordance with RG 1.207 and NUREG/CR-6909. This analysis considers the effects of environmentally assisted fatigue. The staff finds that this complies with SRP Section 3.9.3, Acceptance Criterion II.1, and is, therefore, acceptable.

In its response to **RAI 9021, Question 03.09.03-3(3)**, the applicant stated that 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. Therefore, **RAI 9021, Question 03.09.03-3**, is resolved and closed.

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.

<u>Steam Generator Operability Assurance</u>

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 ASME *Operation and Maintenance of Nuclear Power Plants*, Division 1: Section IST (OM Code).

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 environmental qualification 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, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 3, issued September 2009, 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 issued **RAI 9021, Question 03.09.03-4**, asking the applicant to discuss whether and how the SG meets the operating ability requirements in NUREG-1367, “Functional Capability of Piping Systems,” issued November 1992.

In its response dated October 23, 2017 (ADAMS Accession No. ML17296B367), to **RAI 9021, Question 03.09.03-4**, 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 actually 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. In-structure response spectra 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 1, 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, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” Revision 36, issued August 2014.

The staff finds the conditions provided above meet the conditions in Section 9 of NUREG-1367 with the exception of the external pressure, which is designed in compliance with NRC-accepted ASME Code Case N-759-2 and is therefore acceptable. Therefore, **RAI 9021, Question 03.09.03-4**, is resolved and closed.

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 is complete. The staff noted a lack of stresses and fatigue evaluations for internals and critical representative RPV components. This is considered **Open Item 03.09.03-1** until NuScale provides satisfactory stress data and a fatigue evaluation as they relate to RPV and RVI components. NuScale provided a letter dated February 4, 2019 (ADAMS Accession No. ML19035A682), indicating that the stress analysis for the RPV would be completed by July 2019. NuScale prioritized the portions to be evaluated for fatigue, analyzing the most critical locations by June 2019.

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. 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, "Methodology for Combining Dynamic Responses," Revision 1, 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 Code, Section III, Subsection NF. The core support structures are designed to ASME 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 only applicable to the fatigue analysis. The top support structure 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 top support structure 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 Code, Section III, Subsection NF. The core support structures are designed to ASME 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 issued **RAI 9021, Question 03.09.03-5**, requesting the applicant to discuss 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.

In its response dated October 23, 2017 (ADAMS Accession No. ML17296B367), to **RAI 9021, Question 03.09.03-5**, the applicant stated that TR-0716-50439 discusses the flow-induced vibration 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.

Using the analytically determined root mean square response, a fretting wear assessment is performed 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 turbulent buffeting. The coefficient of friction 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 supplement response dated March 8, 2018 (ADAMS Accession No. ML18067A520), to **RAI 9021, Question 03.09.03-5**, the applicant stated that a zero-to-peak root mean square vibration for sinusoidal whirling of the tube is used.[2] This approximation is equal to the square root of two times the calculated root mean square 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 NEI 97-06, Revision 3, "Steam Generator Program Guidelines" dated January 2011 (ADAMS Accession No. ML111310708). TS 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 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, "Steam Generator Program Guidelines", the staff finds this acceptable. Therefore, **RAI 9021, Question 03.09.03-5 is closed.**

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, the functionality assurance, environmental, and seismic qualification programs that are applied to components are also applied to the associated supports. SRP Section 3.9.3 has an extensive discussion on the snubber as the shock- and vibration-energy-absorbing device for its dynamic characteristics. However, 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. DCA Part 2, Tier 2, Section 3.10.1.2, states that a NuScale equipment qualification record file addresses requirements for active valves and dampers. The structural integrity and operability of the active valve is qualified by a combination of analyses and tests. The staff issued **RAI 9021, Question 03.09.03-6**, requesting the applicant to describe these dampers and their use. In its response dated October 23, 2017 (ADAMS Accession No. ML17296B366), to **RAI 9021, Question 03.09.03-6**, the applicant clarified that the NuScale design does not have snubbers or active dampers. DCA Part 2, Tier 2, Section 3.10.1.2, mentioned active dampers in error. The applicant revised DCA Part 2, Tier 2, Section 3.10.1.2, to remove the reference to active dampers, as shown in the accompanying markup. The staff reviewed the revision and found that the applicant had incorporated the markup in DCA Part 2, Tier 2, and finds the change acceptable. Therefore, **RAI 9021, Question 03.09.03-6**, is resolved and closed.

3.9.3.5 Combined License Information Items

There are no COL information items for this area of review.

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[2] Au-Yang, M.K., "Flow-Induced Vibration of Power and Process Plant Components," ASME Press, New York, NY, 2001, pages 343 and 361.

3.9.3.6 *Conclusion*

Because of the **Open Item** discussed above regarding the lack of stress and fatigue information in the application, the staff is not able to conclude 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. Thus, the staff is not able to conclude that the regulatory requirements are met.

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 GDC 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 anticipated operational occurrences, 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. This review 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 control rod drive 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 control rod drive 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: DCA Part 2, Tier 1, Table 2.1-4, provides the ITAAC for DCA Part 2, Tier 2, Section 3.9.4. Section 14.3 of this SER discusses NuScale ITAAC.

Technical Specifications: TS Section 3.1.4, "Rod Group Alignment Limits," contains surveillance requirements pertinent to the review scope of SRP Section 3.9.4, "Control Rod Drive Systems," Revision 3, 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 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, GDC 2, as it relates to the CRDS, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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 anticipated operational occurrences

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, GDC 29, 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 anticipated operational occurrences

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 documents provide 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 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 a Principal Design Criterion 27, which states the following:

> The reactivity control systems shall be designed to have a combined capability of reliable 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, without margin for stuck rods, provided the probability for a return to power assuming a stuck rod is sufficiently small and specified acceptable fuel design limits for critical heat flux would not be exceeded by the return to power.

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.

The staff issued **RAI 58-8835, Question 03.09.04-1**, dated June 8, 2017 (ADAMS Accession No. ML17160A411), asking the applicant to provide legible and detailed drawings (including component identification, class breaks, and dimensions) for all drawings related to SRP Section 3.9.4 to better describe the design of this system. In its response dated August 7, 2017 (ADAMS Accession No. ML17219A749), to **RAI 58-8835, Question 03.09.04-1**, the applicant provided additional details in DCA Part 2, Tier 2, Section 4.6.1, to better describe the contents of the figures in the DCA Part 2. Furthermore, the applicant updated DCA Part 2, Tier 2, Figure 4.6-5, to define all acronyms and add descriptive detail to DCA Part 2, Tier 2, Sections 3.9.4.1.1 and 4.6.1. The applicant supplemented this response with a letter dated November 30, 2017 (ADAMS Accession No. ML17334B725), to add a figure that had been erroneously excluded from the original response. The applicant incorporated these details into DCA Part 2, Tier 2, Revision 1. Based on the incorporation of these details, **RAI 58-8835, Question 03.09.04-1**, is resolved and closed.

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. However, DCA Part 2, Tier 2, Figure 5.1-1, “NuScale Power Module Major Components,” also illustrates the CRDM support structures, showing a different number of support structures than Figure 4.6-1. Additionally, DCA Part 2, Tier 2, Section 3.9.4, does not discuss these support structures or any other means by which the CRDS is supported, despite discussions in DCA Part 2 about the very long length of the control rod drive shafts when compared to traditional large LWRs. The staff issued **RAI 58-8835, Question 03.09.04-2**, dated June 8, 2017 (ADAMS Accession No. ML17160A411), asking the applicant to explain the support configuration so the staff can make a safety finding for the review area of GDC 1 and 2. In its response, dated August 7, 2017 (ADAMS Accession No. ML17219A749), to **RAI 58-8835,**

Question 03.09.04-2 the applicant described the support configuration and proposed markups to DCA Part 2 to better explain the configuration. The proposed revisions to DCA Part 2 were inconsistent with the narrative provided with the RAI response, so the staff discussed the response with NuScale during a public meeting on September 13, 2017 (ADAMS Accession No. ML17277A328). During this discussion, NuScale committed to provide a voluntary submittal (ADAMS Accession No. ML17334B725) to resolve this inconsistency. The applicant incorporated the details of the supplemented response into DCA Part 2, Tier 2, Revision 1. Based on the incorporation of these details, **RAI 58-8835, Question 03.09.04-2**, is resolved and closed.

DCA Part 2, Tier 2, Section 4.6.2, mentions that a failure modes and effects analysis has evaluated failures of the CRDM but the application does not discuss the results of this analysis. Review of this analysis or a summary of its results permits the staff to ensure that the design has been thoroughly analyzed. The staff issued **RAI 58-8835, Question 03.09.04-3**, dated June 8, 2017 (ADAMS Accession No. ML17160A411), asking the applicant to provide this analysis or describe the postulated failures (both mechanical and electrical) so the staff can make a safety finding for GDC 2, 14, 26, 27, and 29. 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. As this information was made available for the staff's review, and the results of this review are acceptable and documented in the audit report and SER Section 4.6, the staff determined that **RAI 58-8835, Question 03.09.04-3**, is resolved and closed.

The staff reviewed the description of the method of operations and noted that additional detail is required to make a safety finding for GDC 26, 27, and 29. Statements like "coils are energized in the sequence," when describing the stepping process provide an insufficient level of detail to make a determination that the operation sequence does not place the system in a non-fail-safe configuration. The staff issued **RAI 58-8835, Question 03.09.04-4**, dated June 8, 2017 (ADAMS Accession No. ML17160A411), asking the applicant to provide additional detail on the configuration of the latching mechanism (e.g., how many latches per mechanism, redundancies present in function). The staff requested that the applicant include specific language to indicate that the CRA drops fully into the core and that the reactor trip function is achievable during any part of the insertion/withdrawal sequence under all design conditions in the discussion of the reactor trip function. In its response, dated August 7, 2017 (ADAMS Accession No. ML17219A749), to **RAI 58-8835, Question 03.09.04-4,** the applicant provided a description of the latching mechanism and the method of operation, which was later supplemented by letter dated January 19, 2018 (ADAMS Accession No. ML18022A514), after a discussion at a public meeting on September 13, 2017 (ADAMS Accession No. ML17277A328). However, the staff identified an issue with the provided description, which the applicant proposed an acceptable correction for in a future revision to DCA Part 2. Based on the above response, the staff is tracking **RAI 58-8835, Question 03.09.04-4**, as **Confirmatory Item 03.09.04-1**, pending resolution of the description issue in a subsequent version of the DCA Part 2.

DCA Part 2, Tier 2, Section 3.9.4.4, describes preoperational testing to verify that design requirements are met for insertion, withdrawal, and drop times. DCA Part 2, Tier 2, Section 4.2.4.2.3, states that Figure 4.3-23, "Control Rod Position versus Time after Trip," provides the drop time. This figure shows a plot of position versus time but does not show the time at which full insertion is achieved, which is insufficient for the staff to make a safety finding for GDC 26, 27, and 29. The staff issued **RAI 58-8835, Question 03.09.04-6**, dated June 8, 2017 (ADAMS Accession No. ML17160A411), asking the applicant to provide a numerical drop time as well as numerical values for other important operational parameters (e.g., trip delay). In its response, dated August 7, 2017 (ADAMS Accession No. ML17219A7549, to **RAI 58-8835, Question 03.09.04-6**, the applicant provided the information for a numerical drop time and a narrative discussion about the trip delay requirement. The staff finds this response acceptable; therefore, **RAI 58-8835, Question 03.09.04-6**, is resolved and closed.

DCA Part 2, Tier 2, Section 3.9.4.1.1, Revision 0, states that the sensor coil assembly is attached to the rod travel housing (a portion of the RCPB), but it is unclear how this attachment is made. The staff issued **RAI 58-8835, Question 03.09.04-7**, dated June 8, 2017 (ADAMS Accession No. ML17160A411), asking the applicant to specify the means of attachment for the sensor coil and drive coil assemblies to the rest of the CRDM system. In its response, dated August 7, 2017 (ADAMS Accession No. ML17219A749), to **RAI 58-8835, Question 03.09.04-7**, the applicant noted that this was an erroneous statement and that the sensor coil assembly sets on a ledge and is not directly attached to the rod travel housing. The applicant proposed a markup to DCA Part 2 to remove this error, which the staff finds acceptable. These markups were incorporated into DCA Part 2, Revision 1. Therefore, **RAI 58-8835, Question 03.09.04-7**, is resolved and closed.

SRP Section 3.9.4 states that "of particular interest are any new and unique features that have not been used in the past." DCA Part 2, Tier 2, Revision 0, Section 1.5.1.7, states there are two new features: a remote disconnect mechanism and a rod position indication. DCA Part 2, Tier 2, Section 3.9.4.4, lists the unique features as a longer control rod drive shaft and a remote disconnect mechanism. 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. DCA Part 2, Tier 2, Section 3.9.4.1.2, states the following:

> *During operation, the CRA in each control bank are held in place by the control rod drive shafts when the drive coils are energized… When a reactor trip signal occurs, the operating coils are de-energized.*

The staff issued **RAI 58-8835, Question 03.09.04-5**, dated June 8, 2017 (ADAMS Accession No. ML17160A411), asking the applicant to provide additional information about the remote disconnect mechanism and other new and unique features of the CRDM design. In its response, dated August 7, 2017 (ADAMS Accession No. ML17219A749), to **RAI 58-8835, Question 03.09.04-5**, the applicant confirmed that the remote disconnect coil is always deenergized during normal operations and remains in this state during a reactor trip. The applicant clarified the function of the remote disconnect mechanism in a proposed markup to DCA Part 2, Tier 2, Section 3.9.4.1.1. Furthermore, the applicant clarified DCA Part 2, Tier 2, Section 1.5.1.7, that the remote disconnect mechanism and the long control rod drive shaft are the features not common to conventional CRDMs, and that rod position indication was not considered such a feature. The staff finds the additional information on the remote disconnect mechanism and clarification with regard to other new and unique features to be acceptable. The applicant incorporated the proposed markups into of DCA Part 2, Tier 2, Revision 1, although the section numbers within Section 1.5.1 have changed (Section 1.5.1.6 and 1.5.1.7 are those sections applicable to this review). Therefore, **RAI 58-8835, Question 03.09.04-5**, is resolved and closed.

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 reactor coolant pressure boundary are classified as ASME BPV Code, Section III, Subsection NB (Quality Group A) components.

DCA Part 2, Tier 2, Section 3.9.4, discusses meeting design requirements in accordance with 10 CFR 50.55a but not the construction requirements. The staff issued **RAI 58-8835, Question 03.09.04-8**, dated June 8, 2017 (ADAMS Accession No. ML17160A411), asking the applicant to confirm, within DCA Part 2, that construction will be in accordance with the same codes used for the design of the components. The staff also requested the applicant specify the codes and standards used to ensure the satisfaction of 10 CFR 50.55a requirements. For nonpressurized components, the staff requested the applicant discuss the codes and standards used for design and construction and the design margins achieved, allowable stress and deformation limits used, how fatigue is considered, and how these are comparable to other similar designs. In its response, dated August 7, 2017 (ADAMS Accession No. ML17219A749), to **RAI 58-8835, Question 03.09.04-8**, the applicant clarified that construction will be in accordance with the codes used for the design of the components and included a proposed markup of DCA Part 2, Tier 2. The applicant provided further discussion on 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 control rod drive shaft was specifically mentioned as a major nonpressure-retaining component, and the proposed DCA Part 2, Tier 2, markup specifies it will be considered an ASME BPV Code, Subsection NG, component (internal structure). The staff engaged with the applicant on this topic during a public meeting on September 13, 2017 (ADAMS Accession No. ML17277A328), as ASME BPV Code, Subsection NG has less-prescriptive requirements applied to internal structures than for other components in other subsections and leaves the stipulation of many conditions to the Certificate Holder. As a result of this discussion, the staff issued followup **RAI 254-9181, Question 03.09.04-10**, dated October 13, 2017 (ADAMS Accession No. ML17286A819), requesting additional information on the specific stipulation of requirements for the control rod drive shaft. The staff discussed with the applicant its response to **RAI 254 9181, Question 03.09.04-10** (ADAMS Accession No. ML17345A943), at a public meeting held on January 24, 2018, where the applicant requested a followup RAI to clarify the specific requirements requested in DCA Part 2, for the control rod drive shaft requirements. The staff issued **RAI 374-9389, Question 03.09.04-12**, dated February 28, 2018 (ADAMS Accession No. ML18059A709), requesting the specific requirements needed in DCA Part 2 to support the staff’s finding. The applicant’s response, dated April 13, 2018, (ADAMS Accession No. ML18103A190) contained proposed markups to specify the requested requirements in DCA Part 2. The staff considers these proposed markups acceptable to provide a reasonable set of stipulated conditions for the control rod drive shaft to assure it is designed to quality standards commensurate with its safety function. The staff is tracking **RAI 374-9389, Question 03.09.04-12**, as **Confirmatory Item 03.09.04-2**, pending incorporation of the proposed changes in a subsequent version of the DCA Part 2.

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 evaluates 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. DCA Part 2, Tier 2, Revision 0, Section 1.5.1.7, and DCA Part 2, Tier 2, Revision 0, Section 1.5.1.12, contain additional discussion of specific testing. DCA Part 2, Tier 2, Revision 0, Section 1.5.1.7, refers to the CRDM proof testing as completed, but that the information gained from this testing does not create a design basis for the final CRDM. DCA Part 2, Tier 2, Revision 0, Section 1.5.1.12, refers to the control rod assembly drop and control rod drive shaft alignment test as currently underway. A staff safety finding in this area requires additional details on the testing of the CRDS. At a public meeting on May 2, 2017 (ADAMS Accession No. ML17257A081), the staff requested test plans, summaries, and results associated with each element of the prototype testing program for the CRDS in order to support SRP Section 3.9.4, Area of Review I.4.

In order to acquire the necessary details on the testing of the CRDS, the staff initiated an audit of the CRDS testing programs (see ADAMS Accession No. ML17158B428).

The staff conducted the audit to gain a better understanding of the design and testing methods for the CRDS. 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. This now-completed testing verified that the system can perform its safety function under conditions that result in misalignment of the support structures for the CRDS. The staff reviewed the results of this misalignment testing through a followup audit. The staff independently confirmed the endurance testing parameters located in DCA Part 2, Tier 2, Section 3.9.4.4, by reconstructing the parameters from the source analyses but noted a discrepancy between these parameters and other design documentation reviewed during the audit, specifically in the number of SSEs. The staff is tracking this issue through **RAI 257-9156, Question 03.09.04-11**, dated October 13, 2017 (ADAMS Accession No. ML17286B065), which was opened as part of the conclusion of the audit. In its response dated November 21, 2017 (ADAMS Accession No. ML17325B718), to **RAI 257-9156, Question 03.09.04-11**, the applicant corrected this discrepancy in a proposed markup to DCA Part 2, in order to reflect the endurance testing requirement of two SSEs. Based on the proposed markup, the staff is tracking **RAI 257-9156, Question 03.09.04-11**, as **Confirmatory Item 03.09.04-3**, pending incorporation of the markup in a subsequent revision of DCA Part 2. Further discussion of this audit may be found in the audit summary report (ADAMS Accession No. ML17331A357).

The staff held a followup audit of CRDS testing results from September 4–October 25, 2018, conducted in accordance with the followup audit plan (ADAMS Accession No. ML18235A509).

During the followup audit, the staff reviewed documentation and results from the control rod assembly drop and control rod drive shaft alignment testing, which were not available at the time of the original 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. NuScale plans to include a discussion on the completion of the testing program and the associated results. The staff is tracking this as **Confirmatory Item 03.09.04-4** to ensure the discussion is incorporated into a future revision of DCA Part 2.

During review of DCA Part 2, Tier 2, Section 3.9.4, the staff noted several instances where specific wording needed to be be revised to enhance clarity and provide a direct indication of regulatory compliance. The staff issued **RAI 58-8835, Question 03.09.04-9**, dated June 8, 2017 (ADAMS Accession No. ML17160A411), asking the applicant to revise the wording accordingly. In its response, dated August 7, 2017 (ADAMS Accession No. ML17219A749), to **RAI 58-8835, Question 03.09.04-9**, the applicant proposed revisions to DCA Part 2 revisions that provide adequate resolution of these issues. The applicant incorporated these proposed revisions into DCA Part 2, Tier 2, Revision 1. Therefore, **RAI 58-8835, Question 03.09.04-9**, is resolved and closed.

3.9.4.5 Combined License Information Items

There are no COL information items for this area of review.

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 control rod drive 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 anticipated operational occurrences, with adequate margins to assure the system's reactivity control function and with extremely low probability of leakage or gross rupture of the RCBP. SER Section 3.9.3 evaluates the specified design transients, design and service loadings, combination of loads, and resulting stresses and deformations under such loading combinations.

The staff finds that the applicant has designed the CRDS to reliably control reactivity changes under postulated accident conditions, as discussed in GDC 27, since it has been designed to quality standards commensurate with its safety functions. The staff also finds that the applicant has met the requirements of 10 CFR Part 50, Appendix A, GDC 29, with respect to designing the CRDS to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences, 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 concluded that the operability assurance program 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 flow path through the RPV, and the applicant's design criteria. The RVIs serve several functions. They provide support and alignment for the reactor core, a flow path 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 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, "NuScale Power Module."

DCA Part 2, Tier 2: Tier 2 information associated with this section is provided in DCA Part 2, Tier 2, Section 3.9.5, "Reactor Vessel Internals."

DCA Part 2, Tier 2, Section 3.9.5, describes the arrangement of the RVI assembly and the flow path 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 control rod drive (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 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."

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

Technical Reports: The following TRs apply to this area of review and the staff used information in these reports to make the safety findings:

- TR-0716-50439-P, Revision 0, dated December 2016, "NuScale Comprehensive Vibration Assessment Program Technical Report"

- TR-0916-51502-P, Revision 2, dated April 2019, "NuScale Power Module Seismic Analysis"

3.9.5.3 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 that reactor internals be designed to quality standards commensurate with the importance of the safety functions performed.

- 10 CFR Part 50, Appendix A, GDC 2, requires that reactor internals be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform safety functions.

- 10 CFR Part 50, Appendix A, GDC 4, requires that 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.

- 10 CFR Part 50, Appendix A, GDC 10, requires that 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).

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, "Required Load Combinations for Reactor Vessel Internals American Society of Mechanical Engineers Stress Analysis." Section 3.9.3 of the 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. However, 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 RCBP greater than limited local yielding nor sufficient damage to significantly impair core cooling capacity. Therefore, on October 10, 2017, the staff issued **RAI 162-8901, Question 03.09.05-14** (ADAMS Accession No. ML17284A092), requesting the applicant to explain the rationale for using the Level C allowable limit for a Service Level D condition.

In its response to **RAI 162-8901, Question 03.09.05-14** (ADAMS Accession No. ML17284A092), dated October 10, 2017, 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 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.

Based on the response above, the staff finds the applicant's response to **RAI 162-8901, Question 03.09.05-14** (ADAMS Accession No. ML17284A092), acceptable because using a Level C service limit for a Level D event is more conservative and thus provides a higher safety margin. Therefore, the staff determined that **RAI 162-8901, Question 03.09.05-14**, is resolved and closed.

DCA Part 2, Tier 2, Section 3.9.3.1.2, "Load Combinations and Stress Limits," in the subsection "Core Support Structures," states that the SG tube supports are internal supports and are designed to the same criteria as the core support structure. The meaning of this statement was unclear to the staff. Specifically, it was unclear whether the SG tube supports are classified as core support structure and are thus designed to ASME Code, Section III, Subsection NG, or if they are classified as internal structures and are designed using ASME Code, Section III, Subsection NG, as a guide. Therefore, the staff issued **RAI 162-8901, Question 03.09.05-16** (ADAMS Accession No. ML17284A092), requesting the applicant to clarify the classification of the SG tube supports, including their seismic classification.

In its response to **RAI 162-8901, Question 03.09.05-16** (ADAMS Accession No. ML17284A092, dated October 10, 2017, the applicant clarified that the SG tube supports are classified as internal structures and are constructed using ASME Code, Section III, Subsection NG, as a guide. The seismic category of the SG tube supports is seismic Category I.

The staff finds the applicant's response to **RAI 162-8901, Question 03.09.05-16** (ADAMS Accession No. ML17284A092, dated October 10, 2017), acceptable because the SG tube supports do not provide a core support function. Therefore, the classification as internal structures is acceptable, and using ASME Code, Section III, Subsection NG, as a guide meets the criteria in SRP Section 3.9.5 and is acceptable. Therefore, the staff determined that **RAI 162-8901, Question 03.09.05-16**, is resolved and closed. RAI 499-9564, Question 05.04.02.01-15 (ADAMS Accession No. ML18295A787) dated October 22, 2018, provides more details regarding the design of the SG tube supports.

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
- upper riser assembly
- flow diverter
- PZR spray nozzles

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 upper riser assembly, lower riser assembly, and the CSA, respectively. These figures reference multiple RVI components. TR-0716-50439-P, Revision 0, and TR-0916-51502-P, Revision 0, both contain more detailed figures of the RVI assemblies.

However, the applicant did not provide a list of core support structures and 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.

Because NuScale is a first-of-a-kind (FOAK) reactor that has a different RVI design than other PWRs, similar plant experiences have limited applicability for comparison with the NuScale design. Therefore, the staff issued **RAI 162-8901, Question 03.09.05-1** (ADAMS Accession No. ML17284A092), requesting the applicant to provide a complete list and description of all RVI components and identify which are core support structures and which are internal structures, the positioning and securing of these components within the RPV, and the provision for axial and lateral retention and support of these components.

In its response to **RAI 162-8901, Question 03.09.05-1** (ADAMS Accession No. ML17284A092), the applicant provided a general overall description of the major RVI components and their method of securing. 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 the lock plate assemblies. The upper support blocks that are welded to the core barrel retain the two assemblies in the horizontal direction. The CSA mounted to the bottom head of the RPV supplies the primary support to the reactor core.

In addition, 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. Therefore, the staff finds the response to **RAI 162-8901, Question 03.09.05-1** (ADAMS Accession No. ML17284A092), acceptable. Thus, **RAI 162-8901, Question 03.09.05-1**, is considered resolved and closed.

TR-0716-50439-P, Revision 0, provides further detail of the RVI assembly and description of the SG, which wraps around the upper riser assembly, and PZR, which is located above the upper riser assembly. 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 tube sheet, which allows the steam to travel through on the secondary side. This report also gives details of the SG tube inlet flow restrictors and mounting plate, the helical SG tube bundle, and the SG support bars that provide support to maintain the tube bundle structural integrity. However, the staff was unclear as to where the jurisdictional boundary or classification break is for the RVIs relative to the SG, PZR, and RPV. Therefore, the staff issued **RAI 162-8901, Question 03.09.05-2** (ADAMS Accession No. ML17284A092), requesting the applicant to identify in detail the locations where the jurisdiction for components that are categorized as RVIs end and to explain the design code or standard for these components where the transition takes place. For instance, Figure 2-5 “Steam plenum region” of TR-0716-50439-P, Revision 0, shows the steam plenum region. The plenum tube sheet has penetrating holes where the top end of the tube bundles end. The categorization (RVIs or pressure boundary) and design code or standard for the plenum tube sheet design was unclear to the staff.

The same observations apply to other components inside the RPV. In a traditional PWR, all the components inside the RPV are considered RVIs (either core support or internal structure), with the exception of the CRDM, fuel elements, and instrumentation. However, because of the integrated nature of the NuScale design, the staff recognizes that this historical jurisdiction of boundary may no longer be true for the NuScale design. There are components inside the

NuScale RPV that perform functions other than traditional RVI components. For instance, the SG tube support bars and lower tube support cantilevers are such components. The staff issued **RAI 162-8901, Question 03.09.05-2**, requesting the applicant to provide the design code or standard for these components and clearly explain where the jurisdiction is, and for pressure boundary components, to provide the ASME design code or standard at such locations.

In its response to **RAI 162-8901, Question 03.09.05-2** (ADAMS Accession No. ML17284A092), the applicant explained that there are no separate jurisdictional 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 jurisdictionally 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 tube sheets and PZR baffle plate, are designed in accordance with ASME Code, Section III, Subsection NB. The FW plenum, including the tube sheets, is designed in accordance with ASME 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 RPV component and are designed in accordance with ASME Code, Section III, Subsection NB. 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 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 RPCB and therefore are not RVI components. The SG and flow restrictors are jurisdictionally part of the RPV component.

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 or standard for the core support blocks appropriate because they are structural attachments to the RPV.

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. The SG tube supports are designed as internal structures. The staff finds the classification and design code or standard for the SG tube supports appropriate because they do not perform a core support function. **RAI 499-9564, Question 05.04.02.01-15** (ADAMS Accession No. ML18295A787) dated October 22, 2018, provides more details regarding the design of the SG tube supports.

In response to **RAI 162-8901, Question 03.09.05-2** (ADAMS Accession No. ML17284A092), the applicant indicated that the SG flow restrictors are not pressure boundary components and thus not classified as part of reactor internals. Furthermore, the applicant indicated that there is no design code associated with the SG flow restrictors. 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 tube sheets, which form part of the RCPB and are designed to ASME Code, Section III, Subsection NB. Since then, the applicant changed the design code or standard for the SG flow restrictors to ASME Code, Section III, Subsection NC as detailed in RAI 499-9564, Question 05.04.02.01-17 (ADAMS Accession No. ML18295A787) dated October 22, 2018. The qualification of the SG flow restrictors is detailed in Section 05.04.02.01 of this safety evaluation. Based on the information provided, **RAI 162-8901, Question 03.09.05-2**, is resolved and closed.

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 upper riser assembly 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 through the lower and upper risers 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 staff issued **RAI 162-8901, Question 03.09.05-3** (ADAMS Accession No. ML17284A092), requesting additional details of the risers and connections. Specifically, the staff requested the applicant to provide more detailed design information on (1) the upper riser hanger brace and how it is connected to the upper riser assembly, (2) the slip joint and how it is connected to the upper riser assembly, and (3) the upper riser supports that are used to support the ICI guide tubes and CRD shafts.

In addition, DCA Part 2, Tier 2, Section 3.9.5.1, states that the upper riser assembly hangs from the PZR baffle plate. It also states that the upper riser assembly is supported by the RPV integral steam plenum (e.g., below the bottom of the PZR). In **RAI 162-8901, Question 03.09.05-3**, the staff requested the applicant to provide a detailed description, including the point of attachment, of how the upper riser hangs from the PZR baffle plate, as stated in DCA Part 2, Tier 2, Section 3.9.5.1.

In its response to **RAI 162-8901, Question 03.09.05-3** (ADAMS Accession No. ML17284A092), the applicant explained that the upper riser assembly 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 upper riser assembly are classified as internal structures designed to ASME 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 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 ICI guide tubes. These supports are part of the upper riser assembly and are classified as internal structures designed to ASME Code, Section III, Subsection NG.

Based on the applicant's response to **RAI 162-8901, Question 03.09.05-3** (ADAMS Accession No. ML17284A092), on the upper riser assembly, its interface to the lower riser, and the CRD shaft supports, the staff finds the design code or standard for these components as ASME

Code, Section III, Subsection NG, which is appropriate because these components do not perform a core support function. The staff finds that the design code or standard of these components meet GDC 1 because the code or standard provides assurance that these components meet quality standards commensurate with the importance of the safety functions to be performed and are acceptable. Therefore, **RAI 162-8901, Question 03.09.05-3**, is resolved and closed.

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 provide added flexibility in the vertical direction to accommodate circumstances that involve sufficient thermal growth to close the vertical gap between the upper and lower riser assemblies. The staff issued **RAI 162-8901, Question 03.09.05-4** (ADAMS Accession No. ML17284A092), requesting the applicant to provide a detailed design description, including a drawing, of this bellows assembly in the lower portion of the upper riser and explain how it is connected to the upper riser assembly, its classification, and its design code and standard. The staff also requested the applicant to describe the vertical gap that exists between the upper riser assembly and the lower riser assembly and how this vertical gap is affected under normal operating conditions and Level D condition such as an SSE.

Furthermore, if the upper riser assembly is not physically attached to the lower riser assembly, it would mean that the upper riser assembly is attached only to the PZR baffle plate at the top of the upper riser, and thus, the upper riser assembly essentially behaves like a vertical cantilever attached at the top. Therefore, in **RAI 162-8901, Question 03.09.05-4**, the staff requested the applicant to provide a more detailed explanation of this design, and to identify the mechanism that prevents the upper riser assembly from swinging laterally during all service level conditions and how this affects the structural integrity of the ICI guide tubes and the CRD shafts.

In its response to **RAI 162-8901, Question 03.09.05-4** (ADAMS Accession No. ML17284A092), the applicant explained that the bellows is part of the upper riser assembly. The bellows are located a few inches above the cone structure at the base of the upper riser. The upper riser assembly, including the bellows, is classified as an internal structure and is designed to ASME Code, Section III, Subsection NG, as a guide. In the cold condition, the bellows applies approximately 226.8 kilograms (500 lbs) to the lower riser interface. Final design of the bellows is not yet complete. The reference in the DCA Part 2, to a gap related to an earlier design configuration before the leakage flow was evaluated and found not to be acceptable. The referenced gap no longer exists. DCA Part 2, Tier 2 Section 3.9.5.1 has been updated with the current configuration.

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. These cantilever supports limit extreme motion of the upper riser. 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 acceptable because the SG tube support structure limits the movement of the upper riser. Further detail about the SG tube support structure is documented in RAI 499-9564, Question 05.04.02.01-15 (ADAMS Accession No. ML18295A787). The result of the CRD shaft alignment drop test is documented as stated above. In addition, RAI 410-9310, Question 03.09.02-64 (ADAMS

Accession No. ML18201A267 and ML18288A269) provides more details for the design of the bellow. Specifically, the upper and lower lateral restraints of the bellow overlap prevents relative lateral displacement between the upper and lower sections of the bellow assembly. This lateral restraint is necessary in order to provide structural support for the control rod drive shafts and the ICI guide tubes. The lateral restraints are classified as Seismic Category I. Based on the information provided above and the information documented in other sections of this safety evaluation, the staff finds **RAI 162-8901, Question 03.09.05-4** acceptable and is resolved and closed.

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 ICI guide tube 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.

The staff issued **RAI 162-8901, Question 03.09.05-5** (ADAMS Accession No. ML17284A092), requesting additional detail for the CRA guide tube support plate, CRDS support structure, ICI guide tube support, upper core plate, fuel pins, and other structures.

In its response to **RAI 162-8901, Question 03.09.05-5** (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 Code, Section III, Subsection NG.

The CRD shaft support structure is a grid, which is also called the ICI guide tube support. This structure supports both the CRD shaft and ICI guide tube. 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 structure is classified as an internal structure and is designed to ASME 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 Code, Section III, Subsection NG.

The applicant also submitted supplemental response to RAI 162-8901, Question 03.09.05-5 (ADAMS Accession No. ML19008A413) dated January 8, 2019, that also stated that the lock plate assemblies that secure the upper core plate have been replaced with socket head cap screws and alignment dowel pins. RAI 410-9310, Question 03.09.02-62 (ADAMS Accession No. ML18207A525 and ML18305B313) provides more details about this configuration. Based on the information provided by the applicant, the staff finds that the design code or standard for the lower riser assembly meets GDC 1 because the code or standard provides assurance that the lower riser assembly meets quality standards commensurate with the importance of the safety functions to be performed and is acceptable. Therefore, **RAI 162-8901, Question 03.09.05-5**, is resolved and closed.

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, lock plate assembly, 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, Revision 0, briefly describes each of the major components of the CSA.

The staff issued **RAI 162-8901, Question 03.09.05-8** (ADAMS Accession No. ML17284A092), requesting additional information on the core barrel and its interface with other RVIs.

In its response to **RAI 162-8901, Question 03.09.05-8** (ADAMS Accession No. ML17284A092), 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 a cutout to mechanically couple the lower riser to the core barrel via a locking mechanism that is part of the upper support blocks. The core barrel is classified as a core support structure and is designed to ASME 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 2 feet tall and 10 inches wide and fill the space between the core barrel and the RPV. The 10-inch 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 a locking mechanism 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 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 8 inches 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 Code, Section III, Subsection NG. The guide feature is welded to the vessel wall and is part of the vessel. The applicant also provided supplemental information to **RAI 162-8901, Question 03.09.05-8** (ADAMS Accession No. ML18260A227) to clarify the gap between the upper support block and the inside wall of the RPV during both hot and cold conditions.

The applicant also provided information on the lower core plate, core support blocks, reflector blocks, and surveillance specimen capsule holders. 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 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 Code, Section III, Subsection NG.

Four core support blocks are welded to the bottom of head of the RPV. The core support blocks perform a core support function, they are jurisdictionally part of the RPV and are designed to ASME Code, Section III, Subsection NB. 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. **RAI 410-9310, Question 03.09.02-62** (ADAMS Accession No. ML18207A525 and ML18305B313) provides more details about the core support blocks and its 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 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 Code, Section III, Subsection NG. The surveillance specimen capsule holders themselves are classified as internal structures and are designed to ASME Code, Section III, Subsection III, Subsection NG.

Based on the information provided by the applicant, the staff held multiple discussions with the applicant to gain better understanding of the mechanism to secure the core support blocks to the lower core plate, as well as relative movement of the reflector blocks. Both of these issues are discussed in Section 3.9.2 of the safety evaluation. Specifically, RAI 410-9310 RAI Question 03.09.02-62 (ADAMS Accession No. ML18207A525 and ML18305B313) provides more detail for the mechanism to secure the core support blocks to the lower core plate, while RAI 202-8911 Questions 03.09.02-45 (ADAMS Accession No. ML19031C983 and ML19072A151) provides 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 core support assembly and its interface with other RVIs meets GDC 1 and is acceptable. Therefore, **RAI 162-8901, Question 03.09.05-5**, is resolved and closed.

3.9.5.4.1.5 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 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 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, Revision 0, 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 staff issued **RAI 162-8901, Question 03.09.05-6** (ADAMS Accession No. ML17284A092), requesting the applicant to provide more detailed information about the CRA guide tube.

In its response to **RAI 162-8901, Question 03.09.05-6** (ADAMS Accession No. ML17284A092), the applicant stated that 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 Code, Section III, Subsection NG. The CRA has fully withdrawn and fully inserted positions.

Based on the information provided, the staff finds the applicant's response acceptable because the applicant clarified the design of the CRA guide tubes. Therefore, the staff determined that **RAI 162-8901, Question 03.09.05-6**, is resolved and closed.

3.9.5.4.2.1 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 ICI guide tube support structure is located inside the lower riser to support and align ICI guide tubes with their respective fuel assemblies. DCA Part 2, Tier 2, Figure 3.9-3 "Lower Riser Assembly" shows a typical ICI guide tube.

The staff issued **RAI 162-8901, Question 03.09.05-7** (ADAMS Accession No. ML17284A092), requesting the applicant to provide more detailed information on the ICI guide tube.

In its response to **RAI 162-8901, Question 03.09.05-7** (ADAMS Accession No. ML17284A092), the applicant explained that there are 12 ICI guide tubes. Each ICI guide tube 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 ICI guide tube is supported by the five CRD shaft supports. The interface between the ICI guide tube 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 ICI guide tube spans the height of the lower riser. The top end of this segment fits in the socket in the lower side of the ICI guide tube support at the top of the lower riser assembly. The tube is a slip/clearance fit in the ICI guide tube support at this location to allow for thermal expansion. The ICI guide tubes 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 ICI guide tubes. The fourth segment is part of the fuel assemblies. The upper three ICI guide tube segments are classified as internal structures and are designed to ASME Code, Section III, Subsection NG.

The staff finds that the design code or standard of the ICI guide tube meets GDC 1 because the design meets quality standards commensurate with the importance of the safety functions to be performed and is, therefore, acceptable. Therefore, **RAI 162-8901, Question 03.09.05-7**, is resolved and closed.

3.9.5.4.2.2 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.

The staff issued **RAI 162-8901, Question 03.09.05-9** (ADAMS Accession No. ML17284A092), requesting the applicant to provide a detailed design description, including a drawing of the flow diverter, its classification and design code/standard, its attachment and interface, and other details.

In its response to **RAI 162-8901, Question 03.09.05-9** (ADAMS Accession No. ML17284A092), the applicant explained 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 Code, Section III, Subsection NG.

Based on the information provided, the staff finds the applicant's response acceptable because the applicant clarified the design of the flow diverter and that it does not interfere with other core support structures. Therefore, the staff determined that **RAI 162-8901, Question 03.09.05-9**, is resolved and closed.

3.9.5.4.3 Deformation Limit

DCA Part 2, Tier 2, Section 3.9.5, provides no information on deformation limits, such as an acceptable deformation limit for RVIs at which safety function can still be maintained and the justification of that limit. Therefore, the staff issued **RAI 162-8901, Question 03.09.05-15** (ADAMS Accession No. ML17284A092), requesting the applicant to provide information on deformation limits for RVI components under all service-level conditions.

In its response to **RAI 162-8901, Question 03.09.05-15** (ADAMS Accession No. ML17284A092), the applicant explained that the reactor internals components that require deflection limits are the upper riser (including the CRD shaft supports) and the lower riser (including CRA guide tubes and upper core plate). Deflection limits are imposed on these 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 will 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 ADAMS Accession No. ML18325A153. Based on the information provided and the result of the audit, the staff finds the information provided by the applicant meets GDC 1 because it provides quality standards commensurate with the importance of the safety functions to be performed and is acceptable. Therefore, **RAI 162-8901, Question 03.09.05-15**, is resolved and closed.

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 since a main steamline or FW line break can adversely affect the RVI components, and their categorization as a Level D condition is appropriate. Level D evaluation in documented in Section 3.9.2 of the SER.

3.9.5.4.5 Flow-Induced Vibration

Section 3.9.2 of this SER addresses the results of the comprehensive vibration assessment program including the preoperational 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

Since the NuScale reactor is an FOAK reactor with the integrated feature of the SG and PZR, it is essential for the staff to understand the flow pattern within the RPV. One design parameter the staff considers is core bypass flow, the locations at which core bypass flow are expected to occur, and the percentage of the core bypass flow compared to full flow. Therefore, the staff issued **RAI 162-8901, Question 03.09.05-10** (ADAMS Accession No. ML17284A092), requesting the applicant to provide a detailed design description of core bypass flow.

In its response to **RAI 162-8901, Question 03.09.05-10** (ADAMS Accession No. ML17284A092), 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 best estimate bypass flow through the reflector cooling channels is 4 percent of total flow, and through the fuel assembly guide and instrument tubes is 3.3 percent of total flow. DCA Part 2, Tier 2, Table 4.1-1, shows the total best estimate of bypass flow.

The staff finds the information provided by the applicant acceptable because it clarifies the core bypass flow. The adequacy of the core bypass flow is evaluated in Section 4 of this SER. Therefore, **RAI 162-8901, Question 03.09.05-10**, is resolved and closed.

3.9.5.4.6.2 Other Design Parameters—Reactor Vessel Internals Gap Fit

One design parameter that the staff considers in reviewing RVIs is the gap fit and how each RVI component fits to each other during both hot and cold conditions. However, DCA Part 2, Tier 2, Section 3.9.5, provides no information about RVI gap fit during hot and cold conditions. Therefore, the staff issued **RAI 162-8901, Question 03.09.05-11** (ADAMS Accession No. ML17284A092), requesting the applicant to provide detailed design information about RVI components' hot gap and cold gap fit, the effect of thermal expansion of various RVI components, and the potential for any interference.

In its response to **RAI 162-8901, Question 03.09.05-11** (ADAMS Accession No. ML18023B471), 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), Item 3, would provide verification that the ASME BPV Code requirements related to hot gap and cold gap fit for RVIs 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, the issue with RVI components' gap fit is resolved. The staff finds the applicant's response to **RAI 162-8901, Question 03.09.05-11** acceptable, and the question is considered resolved and closed.

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.

The staff issued **RAI 162-8901, Question 03.09.05-12** (ADAMS Accession No. ML17284A092), requesting the applicant to clarify, in the next revision of DCA Part 2, the nomenclature used for upper core plate versus upper core support plate, lower core plate versus lower core support plate, and seismic support information.

In addition, the staff requested the applicant to provide information on how, during seismic and accident conditions, the lateral load is transferred from the core barrel to the RPV shell through the core support blocks, which are located at the bottom of the RPV, and through the upper support blocks, which are located at the upper portion of the core barrel.

In its response to **RAI 162-8901, Question 03.09.05-12** (ADAMS Accession No. ML17284A092), the applicant clarified that the correct nomenclature is upper core plate and lower core plate instead of upper core support plate and lower core support plate. The applicant provided a markup of DCA Part 2, and this correction will be made in the next DCA Part 2 revision.

The applicant also explained that during normal operating conditions, there are no lateral loads transmitted between the core barrel and the RPV. During seismic or blowdown events, lateral loads are transmitted through either or both the upper support blocks and the core support blocks.

The applicant also provided supplemental information to RAI 162-8901, Question 03.09.05-12 (ADAMS Accession No. ML19008A413) 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.

The staff finds the information provided by the applicant acceptable because it clarifies the load path and the mechanism to hold down the core support. The staff also reviewed the markup of DCA Tier 2, Section 3.9.5.1, and confirmed that the markup was incorporated into Revision 1 of DCA Tier 2, Section 3.9.5.1. Therefore, **RAI 162-8901, Question 03.09.05-12**, is resolved and closed.

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 upper riser assembly stays attached to the upper section of the NPM (upper containment vessel (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.

The staff issued **RAI 162-8901, Question 03.09.05-13** (ADAMS Accession No. ML17284A092), requesting additional information on the refueling operation configuration of the upper riser and CNV stand.

In its response to **RAI 162-8901, Question 03.09.05-13** (ADAMS Accession No. ML17284A092), the applicant explained 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 RPV integral steam plenum.

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 ICI guide tubes that protrude below the upper core plate. The lower riser assembly stand is not a safety-related component, and its detailed design has not yet been completed.

The staff finds that the information provided by the applicant acceptable because it clarifies the measure to prevent damage of the fuel pins. Therefore, **RAI 162-8901, Question 03.09.05-13**, is resolved and closed.

The staff conducted the regulatory audits (Phases 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), respectively. The staff finds the RVI design information described in DCA Part 2, Tier 2, was adequately translated into the design specification documentation. However, the staff finds that the mechanical loads for RVI, stress and fatigue analysis which is information required to have a completed design of RVI, are not provided for the staff’s reviews. The staff plans to audit this necessary information in May 2019. Therefore, **Open Item 03.09.03-1** is created to request the applicant to provide the stress and fatigue reports for the staff’s audit.

3.9.5.5 Combined License Information Items

Table 3.9.5-1 lists COL information item numbers and descriptions related to the RVIs from DCA Part 2, Tier 2, Table 1.8-2.

Table 3.9.5-1 NuScale COL Information Items for Section 3.9.5

COL 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

Until **Open Item 03.09.03-1** discussed in section 3.9.3 “ASME BPV Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures” of this SE, which tracks the progress of the stress and fatigue analysis of RVI, is resolved, the staff is unable to reach a conclusion about whether the RVI design meets the regulatory requirements.. The staff has held continued conversations with the applicant to provide the stress and fatigue analysis by July 2019 and an audit is planned to start May 2019.

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 2 of DCA Part 2 on October 30, 2018 (ADAMS Accession No. ML18311A006). 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 states 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, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," as endorsed in RG 1.100, Revision 3, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," with clarifications as described in 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 rupture disk or 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 and DHRS boundary, (3) power-operated valve (POV) tests consisting of the AOVs and HOVs, (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) containment vessel 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 3.9.6.5, “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 and includes valves in the CVCS, CFWS, and MSS.

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, "Pump and Valve Operability Assurance," 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 emergency core cooling system (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. The staff describes its review of ITAAC in Chapter 14 of this SER.

Technical Specifications: Part 4 of the NuScale DCA includes the TS 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 TS 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 feedwater regulation valves (FWRVs).

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:

- 10 CFR Part 50, Appendix A, 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.

- 10 CFR Part 50, Appendix A, 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.

- 10 CFR Part 50, Appendix A, 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.

- 10 CFR Part 50, Appendix A, 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.

- 10 CFR Part 50, Appendix A, 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.

- 10 CFR Part 50, Appendix A, 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.

- 10 CFR Part 50, Appendix A, 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.

- 10 CFR Part 50, Appendix A, 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.

- 10 CFR Part 50, Appendix A, 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.

- 10 CFR Part 50, Appendix A, 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 a final safety analysis report 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.

3.9.6.4 Technical Evaluation

In accordance with 10 CFR Part 52, 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. As part of this review, 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 final safety analysis report, in support of a COL application.

DCA Part 2, Tier 2, Section 3.9.6, summarizes the PST and IST programs to be developed by a COL applicant that references the NuScale DC.

On August 4, 2017, the staff issued **RAI 8952, Question 03.09.06-5** (ADAMS Accession No. ML17216A318), requesting the applicant to describe the intent regarding fully describing the PST and IST programs for reference by a COL applicant, or relying on the COL applicant to supplement DCA Part 2 to fully describe the PST and IST programs consistent with the Commission policy for operational programs for new reactors, as described in SECY-02-0067, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Operational Programs (Programmatic ITAAC)," dated April 15, 2002 (ADAMS Accession No. ML020700641); 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 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).

In a letter dated November 2, 2017 (ADAMS Accession No. ML17306A803), NuScale responded to **RAI 8952, Question 03.09.06-5**, with a proposed revision to DCA Part 2, Tier 2, Section 3.9.6. In its RAI response, NuScale 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. NuScale plans to incorporate the changes specified in the responses to RAI 8952 and additional clarifications in a future revision to DCA Part 2. The remaining clarifications include the following:

- updating the reference list to include the 2017 Edition of the ASME OM Code
- clarifying the description of the function position that is not safety-related for check valves in Section 3.9.6
- updating Note 6 in Table 3.9-15, “Active Valve List,” to specify the augmented function as described in Section 3.9.6 and the accident analyses
- updating the OM Code Category for ECCS valves in Table 3.9-16 based on the final valve design, as necessary
- updating Table 3.9-17 for performance assessment testing as part of the augmented testing program
- updating Table 3.9-17 to include leakage testing for CVC-CKV-0329 and 0323 in the OM Code Category A/C

Therefore, the staff will track **RAI 8952, Question 03.09.06-5**, as **Confirmatory Item 03.09.06-1**, pending submittal of the next revision of DCA Part 2, which will incorporate the RAI responses and clarifications to fully describe the PST and IST programs for the NuScale reactor, 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. Therefore, the staff issued **RAI 8820, Question 03.09.06-3**, dated June 2, 2017 (ADAMS Accession No. ML17153A377), requesting the applicant to describe its basis for the functional capability qualification provisions for safety-related valves for the NuScale Power Plant.

In its response (ADAMS Accession No. ML17213A540), dated August 1, 2017, to **RAI 8820, Question 03.09.06-3**, 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. Subsequently, the staff issued **RAI 9131, Question 14.03.03-6**, on November 3, 2017, requesting the applicant to discuss its plans to specify the qualification process for safety-related valves as part of DCA Part 2, Tier 1. In its responses to **RAI 9131, Question 14.03.03-6**, 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, "Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria," 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. Therefore, the staff has determined that **RAI 8820, Question 03.09.06-3**, is resolved and closed.

The NRC regulations in 10 CFR 52.47 indicate that the NRC will require, before DC, 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 issued **RAI 8820, Question 03.09.06-4** (ADAMS Accession No. ML17153A377), requesting the applicant to provide its schedule for completing the information normally contained in the procurement specifications, and construction and installation specifications, for safety-related valves (such as those in the ECCS and containment isolation system) and for making this information available for audit to allow the NRC to reach a safety determination on the NuScale DCA.

In its response dated August 1, 2017 (ADAMS Accession No. ML17213A540), to **RAI 8820, Question 03.09.06-4**, the applicant stated that the type of information requested is contained in the design specifications for the ECCS valves, which was 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 design specifications from June 1 to August 29, 2017. A report dated January 25, 2018 (ADAMS Accession No. ML18018A234) documents the staff's audit results and comments provided to NuScale. The staff conducted a followup audit of the NuScale design specifications from May 14 through October 11, 2018. In response to the followup audit, NuScale is updating the design specifications and will notify the NRC when the design specifications are available, as appropriate, for the staff to compete its safety review. The staff issued the followup audit report on February 11, 2019 (ADAMS Accession No. ML19018A140). Therefore, the staff will track **RAI 8820, Question 03.09.06-4**, as **Confirmatory Item 03.09.06-2** for the update of the design specifications.

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). A COL applicant planning to implement a certified design must satisfy the NRC regulations related to the IST program for safety-related valves specified in 10 CFR 50.55a.

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. In describing its IST program, a COL applicant referencing the NuScale design must satisfy these NRC regulations specifying the applicable edition and addenda of the ASME OM Code.

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.

On August 12, 2017, the staff issued **RAI 8953, Question 03.09.06-25** (ADAMS Accession No. ML17224A001), requesting the applicant to describe its plans to satisfy requirements for design and accessibility to perform the PST and IST activities specified in the ASME OM Code to demonstrate the operational readiness of safety-related valves to perform their safety functions based on the design of the NuScale Power Plant.

In its response dated October 9, 2017 (ADAMS Accession No. ML17282A008), to **RAI 8953, Question 03.09.06-25**, the applicant stated that it intends to meet the requirements for design and accessibility to perform PST and IST activities as specified in the ASME OM Code and 10 CFR 50.55a. The applicant indicated that all safety-related valves have been designed to allow for the ability to perform PST and IST activities, and no requests for relief from the ASME OM Code with regard to accessibility are anticipated. In addition, 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 **RAI 8953, Question 03.09.06-25**, is resolved and closed.

DCA Part 2, Tier 2, Section 3.9.6.3, provides a general description of the IST program for valves in the NuScale Power Plant. DCA Part 2, Tier 2, Section 3.9.6.3, and other sections indicate that valves will be grouped for analysis or testing in accordance with the ASME OM Code. The 2012 Edition of the ASME OM Code includes specific provisions for valve grouping of MOVs, safety and relief valves, and check valves. On August 5, 2017, the staff issued **RAI 8955, Question 03.09.06-13** (ADAMS Accession No. ML17217A020), requesting the applicant to describe the plans to group other valve types in accordance with the ASME OM Code. In its response dated October 5, 2017 (ADAMS Accession No. ML17278A999), to **RAI 8955, Question 03.09.06-13**, the applicant stated that there are no plans to group other valve types that are not already included or identified in the ASME OM Code as all NuScale Power Plant valve types are addressed by the Code. The applicant stated that check valves and safety and relief valves are grouped in accordance with the ASME OM Code. DCA Part 2, Tier 2, Section 3.9.6.3.2, describes the grouping of check valves and safety and relief valves in accordance with the ASME OM Code. Therefore, the staff has determined that **RAI 8955, Question 03.09.06-13**, is resolved and closed.

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, such as AOVs, HOVs, and solenoid-operated valves (SOVs), in the NuScale Power Plant provided in DCA Part 2, Tier 2, Section 3.9.6. 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.

The initial submittal of DCA Part 2, Tier 2, Section 3.9.6, did not provide a full description of the IST program for POVs in the NuScale Power Plant. Therefore, on August 5, 2017, the staff issued **RAI 8954, Question 03.09.06-12** (ADAMS Accession No. ML17217A019), requesting the applicant to provide a full description of the IST program for POVs for a NuScale Power Plant, or to specify a COL action item regarding POV testing. In its response dated November 2, 2017 (ADAMS Accession No. ML17306A754) to **RAI 8954, Question 03.09.06-12**, NuScale proposed a revision to DCA Part 2, Tier 2, Section 3.9.6, similar to its responses to **RAI 8952, Question 03.09.06-5**, and **RAI 8955, Question 03.09.06-16**. Based on followup discussions, the applicant submitted a supplemental response, dated February 19, 2018 (ADAMS Accession No. ML18050A053), to **RAI 8955, Question 03.09.06-16**, which provided a proposed revision to Section 3.9.6 specifying that POVs will be tested in accordance with Appendix IV to the ASME OM Code (2017 Edition). The proposed revision to Section 3.9.6 included a request to apply Appendix IV to the ASME OM Code (2017 Edition) as an alternative to the 2012 Edition incorporated by reference in 10 CFR 50.55a. As a replacement for quarterly stroke-time testing of AOVs in the 2012 Edition of the ASME OM Code, Appendix IV to the 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. 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 incorporates the lessons learned for POV performance discussed in RIS 2000-03. The staff discusses its review and acceptance of the alternative request submitted by NuScale later in this SER section. The staff has confirmed that DCA Part 2 includes the applicant's response to **RAI 8954, Question 03.09.06-12**. Therefore, **RAI 8954, Question 03.09.06-12**, is resolved and closed. For other types of POVs, such as SOVs, DCA Part 2, Tier 2, Section 3.9.6, specifies that the provisions of the ASME OM Code will be applied in the IST program for the NuScale Power Plant. The staff finds that the description of the IST provisions in the ASME OM Code, as incorporated by reference in 10 CFR 50.55a, for the remaining types of POVs (such as SOVs) in the NuScale Power Plant satisfies the NRC regulations and, therefore, is acceptable.

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. This section also specifies that the check valves will be exercised to both the open and closed positions regardless of their safety function position. Table 3.9-16 identifies the NuScale check valve test frequencies. Section 3.9.6.3.2 indicates that the check valves will be grouped when applying the ASME OM Code, Appendix II. 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 (PIVs) 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 issued **RAI 8955, Question 03.09.06-14** (ADAMS Accession No. ML17217A020), requesting the applicant to describe the design aspects of the NuScale Power Plant that eliminate the need for PIVs to isolate the RCS or containment vessel. In its response dated October 5, 2017 (ADAMS Accession No. ML17278A999) to **RAI 8955, Question 03.09.06-14**, the applicant stated that the NuScale Power Plant design does not use PIVs to isolate either the RCS or the containment vessel (CNV). The applicant indicated that pressure isolation is performed by CIVs and summarized the RCS isolation functions consistent with the DCA Part 2 description. The staff finds the applicant's description of the function of the CIVs to provide pressure isolation to be acceptable for the NuScale IST program. Therefore, the staff has determined that **RAI 8955, Question 03.09.06-14**, is resolved and closed.

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, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," and paragraph ISTC-3620, "Containment Isolation Valves," of the ASME OM Code. The staff describes its review of the IST tables, including CIV leak-testing provisions, later in this SER section.

3.9.6.4.3.6 Inservice Testing Program for Safety and Relief Valves

Paragraph (5), "Pressure Relief Device Tests," in DCA Part 2, Tier 2, Section 3.9.6.3.2, specifies that the PST and IST provisions for the pressure relief devices are identified in Appendix I to the ASME OM Code. The staff issued **RAI 8955, Question 03.09.06-15** (ADAMS Accession No. ML17217A020), requesting the applicant to clarify, in the initial submittal of DCA Part 2, whether the grouping of the safety and relief valves in the NuScale Power Plant design will satisfy the provisions of Appendix I to the applicable edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a. In its response dated October 5, 2017 (ADAMS Accession No. ML17278A999), to **RAI 8955, Question 03.09.06-15**, the applicant stated that the safety and relief valves included in the IST program will meet the requirements of the ASME OM Code, Appendix I, and provided a planned modification to clarify the DCA Part 2 language. The staff finds that the planned modification of DCA Part 2, Tier 2, Section 3.9.6, would clarify the provision for the NuScale safety and relief valves to meet the requirements of the ASME OM Code, Appendix I, as incorporated by reference in 10 CFR 50.55a. The staff has confirmed that paragraph (5) in DCA Part 2, Tier 2, Section 3.9.6.3.2, specifies that the preservice and periodic inservice tests for pressure relief devices will be performed as required by 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, **RAI 8955, Question 03.09.06-15**, is resolved and closed.

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 Inservice Testing Program Tables

The initial submittal of DCA Part 2, Tier 2, Section 3.9.6, included tables with examples of the IST program planned for various types of valves in the NuScale Power Plant. The staff reviewed the IST program tables in Section 3.9.6 to determine whether DCA Part 2 fully described the PST and IST program for the NuScale Power Plant in accordance with 10 CFR 50.55a. Based on the RAIs identified in this SER section and discussions with NuScale personnel, the applicant revised DCA Part 2 to provide a full description of the PST and IST program for the NuScale Power Plant, including a revision of the IST 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. In the following paragraphs, the staff indicates the remaining items to be completed for its review of the IST program tables for the NuScale Power Plant.

On August 5, 2017, the staff issued **RAI 8956, Question 03.09.06-17** (ADAMS Accession No. ML17217A021), requesting the applicant to describe the plans to satisfy the ASME OM Code provisions for preservice testing, or the plans to request relief from or an alternative to the ASME OM Code provisions in accordance with 10 CFR 50.55a(z). In its response dated October 5, 2017 (ADAMS Accession No. ML17278B221), to **RAI 8956, Question 03.09.06-17**, the applicant stated that a COL applicant may choose to perform preservice testing at the factory. Based on followup discussions, NuScale revised DCA Part 2, Tier 2, Section 3.9.6, to remove the specification of factory preservice testing. NuScale also plans to clarify the discussion of factory testing in DCA Part 2, Tier 2, Section 3.9.3.3, in the next revision of DCA Part 2. Therefore, the staff will track the resolution of **RAI 8956, Question 03.09.06-17**, as **Confirmatory Item 03.09.06-3**, pending revision of DCA Part 2 to remove the reference to factory testing in Section 3.9.3.3.

The staff issued **RAI 8956, Question 03.09.06-18** (ADAMS Accession No. ML17217A021), requesting the applicant to describe the basis for leakage categorization of valves to satisfy the provisions of the ASME OM Code. In its October 5, 2017, response to that RAI (ADAMS Accession No. ML17278B221), the applicant stated that it would revise the basis for leakage categorization of valves to be consistent with paragraph ISTC-1300, "Valve Categories," in the ASME OM Code. Subsequent to this RAI response, the applicant revised the IST tables in DCA Part 2 to not include the valve category definition. However, the staff has confirmed that DCA Part 2, Tier 2, Section 3.9.6.3.1, specifies the valve categories consistent with the ASME OM Code. Therefore, **RAI 8956, Question 03.09.06-18**, is resolved and closed.

The staff issued **RAI 8956, Question 03.09.06-19** (ADAMS Accession No. ML17217A021), requesting the applicant to describe the intended application of its Exercise Test category. In its October 5, 2017, response to that RAI (ADAMS Accession No. ML17278B221), the applicant stated that the intent of the exercise test is to provide reasonable assurance that the valve obturator will move to its safety-related position when called upon. Subsequent to this RAI response, the applicant revised the IST tables in DCA Part 2 to not specify the intent of the exercise test category. However, the staff has confirmed that DCA Part 2, Tier 2, Section 3.9.6.3.2, clarifies the exercise testing for safety-related valves. Therefore, **RAI 8956, Question 03.09.06-19**, is resolved and closed.

The staff issued **RAI 8956, Question 03.09.06-20** (ADAMS Accession No. ML17217A021), requesting the applicant to describe how the leak-testing provisions and frequency specified in the ASME OM Code will be satisfied. In its October 5, 2017, response to that RAI (ADAMS Accession No. ML17278B221), the applicant stated that the leak-testing category of certain valves had been inadvertently categorized and discussed plans to update the categorization. Subsequent to this RAI response, the applicant revised the IST tables to clarify the valve leak testing. In addition, the staff has confirmed that DCA Part 2, Tier 2, Section 3.9.6.3.2, describes the valve leak testing for safety-related valves. Therefore, **RAI 8956, Question 03.09.06-20**, is resolved and closed.

The staff issued **RAI 8956, Question 03.09.06-21** (ADAMS Accession No. ML17217A021), requesting the applicant to clarify the implementation of paragraph ISTC-3550, “Valves in Regular Use,” in the ASME OM Code. In its October 5, 2017, response to that RAI (ADAMS Accession No. ML17278B221), the applicant stated that Back-up Feedwater Check Valves FW-CKV-1007 and FW-CKV-2007 operate in the normal course of reactor operation and provided proposed changes to resolve this question in DCA Part 2, Tier 2, Section 3.9.6. The staff has verified that DCA Part 2 incorporated the planned changes to specify the application of paragraph ISTC-3550 for valves in regular use, including observation of valve performance. Therefore, **RAI 8956, Question 03.09.06-21**, is resolved and closed.

The staff issued **RAI 8956, Question 03.09.06-22** (ADAMS Accession No. ML17217A021), requesting the applicant to describe the basis for the OM category, exercise testing, and safety function for the Decay Heat Removal (DHR) valves. In its October 5, 2017, response to that RAI (ADAMS Accession No. ML17278B221), the applicant stated that the DHR Actuation Valves are classified as ASME OM Code Category B. The staff has confirmed that DCA Part 2, Tier 2, Table 3.9-16, “Valve Inservice Test Requirements per ASME OM Code,” specifies the DHR Actuation Valves as ASME OM Code Category B. Therefore, **RAI 8956, Question 03.09.06-22**, is resolved and closed.

The staff issued **RAI 8956, Question 03.09.06-23** (ADAMS Accession No. ML17217A021), requesting the applicant to describe how the leaktight integrity of each RVV and RRV (including their four valve components) will be demonstrated to satisfy the provisions of the ASME OM Code. In its October 5, 2017, response to that RAI (ADAMS Accession No. ML17278B221), the applicant stated that the RVVs and RRVs do not have specific leakage criteria. The staff has confirmed that Note 12 in DCA Part 2, Tier 2, Table 3.9-16, specifies that the RVVs and RRVs do not have specific leakage criteria, and that seat tightness will be verified in accordance with ASME OM Code, Appendix I. Therefore, **RAI 8956, Question 03.09.06-23**, is resolved and closed.

The staff issued **RAI 8956, Question 03.09.06-24** (ADAMS Accession No. ML17217A021), requesting the applicant to describe the RSV leakage testing provisions. In its response dated

October 5, 2017, to **RAI 8956, Question 03.09.06-24** (ADAMS Accession No. ML17278B221), the applicant stated that seat tightness of the RSVs shall be in accordance with the requirements of the ASME OM Code, Appendix I. The staff has confirmed that Note 13 in DCA Part 2, Tier 2, Table 3.9-16, specifies that the seat tightness of the RSVs will be verified in accordance with ASME OM Code, Appendix I. Therefore, **RAI 8956, Question 03.09.06-24**, is resolved and closed.

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 initial submittal of DCA Part 2 did not include requests for relief from, or alternatives to, the ASME OM Code for the NuScale IST program. The staff issued **RAI 8955, Question 03.09.06-16** (ADAMS Accession No. ML17217A020), requesting the applicant to describe its plans to satisfy requirements to perform the PST and IST activities specified in the ASME OM Code without requests for relief or alternatives to the ASME OM Code in light of the differences in terminology and operating conditions for the NuScale Power Plant design.

In its February 19, 2018, response (ADAMS Accession No. ML18050A053), the applicant submitted a request for relief from the 2012 Edition to the ASME OM Code to define that NuScale Mode 3 "safe shutdown with all reactor coolant temperatures <200 °F" meets the definition of "cold shutdown outage" as defined in paragraph ISTA-2000 of the ASME OM Code (2017 Edition). The applicant stated that the NuScale TS do not include a mode defined as "cold shutdown" as used in the ASME OM Code (2012 Edition).

The NRC regulations in 10 CFR 50.55a(f)(6) state that the Commission will evaluate determinations by an applicant or licensee that specific Code requirements are impractical for a facility. In 10 CFR 50.55a(f)(6), the regulations indicate that the Commission may grant relief and may impose such alternative requirements as it determines 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.

To meet the intent of the definition for "cold shutdown outage" in the 2017 Edition of the ASME OM Code, the applicant proposed that "safe shutdown with reactor coolant temperature <200 °F" is an equivalent condition where the NPM is stable, important safety systems are not required, and cold shutdown testing can commence according to the ASME OM Code requirements.

Subsequent to the RAI response, the applicant updated DCA Part 2, Tier 2, Section 3.9.6.4.1, to describe the relief request in more detail. For example, Section 3.9.6.4.1 specifies that the NuScale TS do not include a Mode defined as "cold shutdown" as used in the ASME OM Code. In particular, Table 1.1-1, "Modes," in 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). 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 420 degrees F. The applicant also stated that containment and containment isolation operability are required at temperatures greater than or equal to 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°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 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, the staff concludes that the “cold shutdown outage” relief request proposed in DCA Part 2, Tier 2, Section 3.9.6.4.1, satisfies 10 CFR 50.55a(f)(6) in that 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. The staff technical review of the NuScale TS is described in Chapter 16 of this SER.

In its response dated February 19, 2018 (ADAMS Accession No. ML18050A053), NuScale submitted a request to implement an alternative to the 2012 Edition of the ASME OM Code in accordance with 10 CFR 50.55a(z). The NRC regulations in 10 CFR 50.55a(z) allow 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.

In its alternative request, the applicant proposed that Appendix IV to the 2017 Edition of the ASME OM Code be applied to the POVs in the NuScale IST Program. The applicant stated that Appendix IV provides an acceptable level of quality and safety by using established ASME OM Code requirements to demonstrate that POVs can perform their safety function under design-basis conditions. Subsequently, DCA Part 2, Tier 2, Section 3.9.6.4.2, incorporated this justification for the alternative request. 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 staff finds that the request 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. 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 is acceptable as it provides an acceptable level of quality and safety in satisfying the requirements in 10 CFR 50.55a(z).

Based on the above evaluation, the staff finds that **RAI 8955, Question 03.09.06-16**, is resolved and closed.

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, “Emergency Core Cooling System,” describes the ECCS, which provides core cooling during and after AOOs and postulated accidents, including design and operation of the RVVs and RRVs. 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 loss-of-coolant accident (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 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 and assisted by a small spring force when the main valve control chamber is vented by tubing through the inadvertent actuation block (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 block 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 block valve 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, 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). 10 CFR 50.43(e), “Additional Standards and Provisions Affecting Class 103 Licenses and Certifications for Commercial Power,” requires that applications for a DC that use 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 DC review of the FOAK design of the NuScale ECCS valves, the staff issued **RAI 8820, Question 03.09.06-1** (ADAMS Accession No. ML17153A377), requesting information on the NuScale ECCS valves (or the schedule for its availability) either in DCA Part 2 or in documentation for NRC audit. The applicant responded to **RAI 8820, Question 03.09.06-**1, in a letter dated August 1, 2017 (ADAMS Accession No. ML17213A540).

Based on the RAI response, the staff conducted multiple audits of the design documentation for the ECCS valves to address this question.

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) have been satisfied. The staff reviewed ECCS valve design documentation made available by NuScale in its electronic reading room. 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 (ADAMS Accession No. ML18052A079), the staff issued a report 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 ventline 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 made available in the NuScale electronic reading room, 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.

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 (POC) 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 reactor 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.

Following completion of the ECCS valve design audit, the staff discussed the audit findings with NuScale to determine the most efficient method to address the followup items for demonstrating the design of the NuScale ECCS valves to support the completion of the NRC review of the NuScale DCA. On September 21, 2018, NuScale submitted its response (ADAMS Accession No. ML18264A312) to the ECCS valve audit report, including ECCS Valve Design Demonstration Testing plans, to resolve the findings from the NRC audit of the ECCS valve design. The staff finds that the NuScale response to the ECCS valve audit report provides an acceptable approach to address the audit findings. Therefore, the staff will track the resolution of **RAI 8820, Question 03.09.06-1**, as **Open Item 03.09.06-1**, based on the plans to address the ECCS valve audit followup items specified in the NuScale letter dated September 21, 2018.

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 issued **RAI 8820, Question 03.09.06-2** (ADAMS Accession No. ML17153A377), requesting the applicant for information on the NuScale CIVs (or the schedule for its availability) either in DCA Part 2 or in documentation for NRC audit. The applicant responded to this RAI in a letter dated August 1, 2017 (ADAMS Accession No. ML17213A540).

Based on the NuScale response to **RAI 8820, Question 03.09.06-2**, 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 include, 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.

At the conclusion of the CIV audit, NuScale agreed to implement the audit findings and submit a letter to the NRC to document its agreement in response to the audit report. On January 31, 2019 (ADAMS Accession No. ML19031C973), NuScale provided its closure plan for the CIV audit followup items, including updates to DCA Part 2 and the design documentation. Once notified that the followup items from the CIV audit have been completed, the staff will review the completed followup items to the extent necessary and finalize its safety evaluation related to the CIV design to support the NRC review of the NuScale DCA. Therefore, the staff will track the resolution of **RAI 8820, Question 03.09.06-**2, as **Confirmatory Item 03.09.06-4**, pending notification by NuScale that the audit followup items have been completed.

DCA Part 2, Tier 2, Figure 6.2-7, "Containment Isolation Valve Actuator Hydraulic Schematic," indicates a rack and pinion arrangement for the operation of the CIVs by a gas bottle actuator. Section 20.1.2.2, "Applicable Structures, Systems, and Components," in Chapter 20, "Mitigation of Beyond-Design-Basis Events," states that the CIVs fail-safe to their closed position using stored energy and references Section 6.2.4 for details of the CIV design and function. During discussions of the NuScale reactor response to a long-term loss of alternating current power, NuScale personnel stated that the CIVs are assumed to remain closed for their design-basis and beyond-design-basis functions. On July 18, 2018, the staff issued **RAI 9565, Question 03.09.06-28**, to request that DCA Part 2, Tier 2, Section 6.2.4, be clarified to specify that each ball valve in the PSCIVs and SSCIVs will be designed and qualified for torque closure using the gas bottle actuator to provide sufficient wedging and sealing to prevent reopening and unsealing of each ball valve following depletion of the gas bottle actuator for the extended time period for the design-basis and beyond-design-basis functions assumed for each individual ball valve.

On September 17, 2018, NuScale submitted its response (ADAMS Accession No. ML18260A299) to **RAI 9565, Question 03.09.06-28**, regarding the design and operation of the CIVs. The staff reviewed the NuScale response to that RAI as part of the CIV audit. At the conclusion of the audit, NuScale agreed to implement the audit findings and to submit a letter to the NRC documenting its agreement with the audit report. Therefore, the staff will track the resolution of **RAI 9565, Question 03.09.06-28**, as **Confirmatory Item 03.09.06-5**, pending notification by NuScale that the audit followup items have been completed.

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.” On August 5, 2017, the staff issued **RAI 8957, Question 03.09.06-8** (ADAMS Accession No. ML17217A014), requesting the applicant to describe the detailed design of the RSVs sufficiently to satisfy the requirements in the ASME Code of record as incorporated by reference in 10 CFR 50.55a with regulatory conditions. In its October 4, 2017, response (ADAMS Accession No. ML17277B826) to that RAI, the applicant stated that the capacity certification for saturated steam at the accumulation pressure provides a means to bound the RSV performance for fluid conditions that may be experienced over the full range of operating conditions, including design-basis accidents. The applicant stated that the design drawings and an ASME design specification for the RSVs were available for audit.

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 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 of the RSV audit include, for example, updates to (1) the RSV design specification to incorporate 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.

At the conclusion of the RSV audit, NuScale agreed to implement the audit findings and submit a letter to the NRC to document its agreement in response to the audit report. 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. NuScale will notify the staff when the followup items from the RSV audit have been completed. The staff will review the completed followup items to the extent necessary to finalize its safety evaluation related to the RSV design to support the NRC review of the NuScale DCA. Therefore, the staff will track the resolution of **RAI 8957, Question 03.09.06-8**, as **Confirmatory Item 03.09.06-6**.

The initial submittal for DCA Part 2, Tier 2, Section 5.2.2.2.2, included COL Item 5.2-2 regarding an 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. The staff issued **RAI 8957, Question 03.09.06-6** (ADAMS Accession No. ML17217A014), requesting the applicant to clarify the COL information item for the capability of the applicable safety and relief valves to perform their safety functions over the full range of fluid flow, differential pressure, and temperature conditions. In its October 4, 2017, response (ADAMS Accession No. ML17277B826) to that RAI, the applicant stated that it would revise COL Item 5.2-2 and DCA Part 2, Tier 2, Table 1.8-2, to clarify that the scope of the Overpressure Protection Report includes low-temperature overpressure protection. The staff found the proposed modifications to COL Item 5.2-2 and DCA Part 2, Tier 2, Table 1.8-2, to be acceptable in clarifying the scope of the Overpressure Protection Report. The staff has confirmed that DCA Part 2 has incorporated the planned changes to revise COL Item 5.2-2 and DCA Part 2, Tier 2, Table 1.8-2, to clarify 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 and NC-7200, to demonstrate that the RCPB and secondary system are designed with adequate overpressure protection features, including low-temperature overpressure protection features. Therefore, **RAI 8957, Question 03.09.06-6**, is resolved and closed.

DCA Part 2, Tier 2, Section 5.2.2.4.1 states 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. The staff issued **RAI 8957, Question 03.09.06-7** (ADAMS Accession No. ML17217A014), requesting the applicant to discuss this provision in DCA Part 2. In its October 4, 2017, response (ADAMS Accession No. ML17277B826) to that RAI, the applicant stated that no credible single active failure can prevent the RSVs from opening and performing their safety function. The applicant stated that the valves open using RCS pressure as the motive force, and therefore, that motive force is always available when RCS overpressurization protection is required. The applicant noted that the RSVs do not require electrical power or external signal to initiate actuation. 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. With respect to the IST program description, **RAI 8957, Question 03.09.06-7**, is resolved and closed.

The initial submittal of DCA Part 2, Tier 2, Section 5.2.2.4.1, stated that the blowdown of each RSV is set greater than 5 percent below set pressure as a deviation from the ASME BPV Code, Section III, Subparagraph NB-7522.6, “Blowdown.” The staff issued **RAI 8957, Question 03.09.06-9** (ADAMS Accession No. ML17217A014), requesting the applicant to provide justification for relief from, or an alternative to, the applicable requirements of the ASME BPV Code, as incorporated by reference in 10 CFR 50.55a for RSV blowdown. In its October 4, 2017, response (ADAMS Accession No. ML17277B826) to that RAI, the applicant stated that it removed the DCA Part 2 statement in Section 5.2.2.4.1 that “each RSV is set greater than 5 percent below set pressure and is a deviation from the ASME BPV Code, Section III, Subparagraph NB-7522.6” from DCA Part 2, as part of the response to **RAI 8914, Question 05.02.01.01-6**. The staff confirmed that this change has been made in DCA Part 2. Therefore, **RAI 8957, Question 03.09.06-9**, is resolved and closed.

The initial submittal of DCA Part 2, Tier 2, Section 5.2.2.9, listed specific tests that would be performed on an RSV prototype to verify RSV performance and reliability. The staff issued

RAI 8957, Question 03.09.06-10 (ADAMS Accession No. ML17217A014), requesting the applicant to describe the test plans for the RSVs to satisfy the provisions of the ASME BPV Code, as incorporated by reference in 10 CFR 50.55a, and ASME QME-1-2007, as accepted in RG 1.100 (Revision 3). In its October 4, 2017, response (ADAMS Accession No. ML17277B826) to that RAI, 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, "Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment," with the specific details addressed in the QME-1 Qualification Specification and QME-1 Qualification Program, and documented in the QME-1 Qualification Report. The applicant indicated that the specific tests listed in DCA Part 2, Tier 2, Section 5.2.2.9, would be deleted. The staff has confirmed that the applicant revised Section 5.2.2.9 as indicated in the RAI response. Therefore, **RAI 8957, Question 03.09.06-10**, is resolved and closed.

The initial submittal of DCA Part 2, Tier 2, Section 5.2.2.10, "Testing and Inspection," included a summary of planned testing of each RSV to be performed by the supplier. The staff issued **RAI 8957, Question 03.09.06-11** (ADAMS Accession No. ML17217A014), requesting the applicant to describe the basis for the summary of the RSV testing provided in DCA Part 2 and to describe the plans for the RSVs to satisfy 10 CFR 50.34(f)(2)(x), which requires an applicant to provide a test program and associated model development and to conduct tests to qualify the RCS relief and safety valves for all fluid conditions expected under operating conditions, transients, and accidents, including consideration of anticipated transients without scram (ATWS) conditions. In its October 4, 2017, response to **RAI 8957, Question 03.09.06-11** (ADAMS Accession No. ML17277B826), the applicant stated that the RSVs will be functionally qualified according to ASME Standard QME-1, as described in DCA Part 2, Tier 2, Section 3.10. The applicant indicated that the details of the qualification will be addressed in the QME-1 Qualification Specification and the QME-1 Qualification Program and documented in the QME-1 Qualification Report. The applicant stated that the qualification program will include testing for all fluid conditions expected under operating conditions, transients and accidents, as required by 10 CFR 50.34(f)(2)(x), including ATWS conditions. The applicant indicated that the specific tests detailed in DCA Part 2, Tier 2, Section 5.2.2.10, would be deleted. On April 11, 2019, the applicant submitted a supplemental response to **RAI 8957, Question 03.09.06-11** (ADAMS Accession No. ML19101A448) proposing to remove the reference to ATWS conditions in Section 3.9.6, based on an ATWS exemption request. The staff will confirm that DCA Part 2 incorporates the planned changes to specify that the RSVs will be qualified in accordance with ASME QME-1-2007, as accepted in RG 1.100 (Revision 3) and as specified in DCA Part 2, Tier 2, Section 3.9.6.1, following completion of the ATWS exemption request review. Therefore, **RAI 8957, Question 03.09.06-11**, will be tracked as **Confirmatory Item 03.09.06-7**.

3.9.6.4.7 Augmented Testing Program

The NRC specified 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 SRMs, 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.

On August 12, 2017, the staff issued **RAI 8958, Question 03.09.06-26**, requesting the applicant to describe the provisions to be applied to provide assurance of the capability of those pumps and valves to perform their intended functions. The staff also requested the applicant to describe the augmented IST program for the applicable pumps and valves to clarify the statement in DCA Part 2, Tier 2, Section 3.9.6, that the components will be tested to the intent of the ASME OM Code.

In its October 11, 2017, response (ADAMS Accession No. ML17284A913) to **RAI 8958, Question 03.09.06-26**, the applicant stated that there are no active pumps or valves that are within the scope of the Commission papers (SECY-93-087, SECY-94-084, and SECY-95-132) for active systems that are not safety-related that provide the first line of defense for the passive emergency cooling system in the NuScale design. Based on followup considerations, NuScale has included DCA Part 2, Tier 2, Section 3.9.6.5 and Table 3.9-16, to describe the 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. DCA Part 2, Tier 2, Table 3.9-17, includes the augmented test requirements for valves in the CVCS, CFWS, CNTS, and MSS.

In addition, NuScale submitted a response dated July 9, 2018 (ADAMS Accession No. ML18190A509) to **RAI 9420, Question 15-17**, dated May 10, 2018, describing its augmented program for valves that are not safety-related relied on in the accident analyses. In a telephone conference on October 23, 2018, NuScale agreed to supplement its response to **RAI 9420, Question 15-17**, to revise DCA Part 2 to justify its reliance on valves that are not safety-related in the NuScale accident analyses. On November 20, 2018, NuScale submitted its supplemental response (ADAMS Accession No. ML18324A889) to **RAI 9420, Question 15-17**. The staff has determined that NuScale's combined responses to this RAI 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. This is consistent with the guidance in 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).

In its RAI responses, NuScale indicated that it will revise DCA Part 2 to specify that these valves that are not safety-related (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. NuScale will also update DCA Part 2, Tier 2, Table 3.2-1, “Classification of Structures, Systems, and Components,” 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 these valves that are not safety-related.

The staff will track **RAI 8958, Question 03.09.06-26**, and **RAI 9420, Question 15-17**, as **Confirmatory Items 03.09.06-8 and 03.09.06-9**, respectively, pending implementation of the planned DCA Part 2 changes.

3.9.6.4.8 Inspections, Tests, Analyses, and Acceptance Criteria

DCA Part 2, Tier 1, specifies ITAAC for safety-related valves in the NuScale Power Plant.

The staff describes its review of ITAAC related to safety-related valves in the NuScale Power Plant in Chapter 14 of this SER. The staff will track the resolution of these RAIs related to the NuScale ITAAC as an **Open Item** as addressed in Chapter 14 of this SER.

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 ASME OM 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 ASME 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 ASME Code Cases listed in Regulatory Guide 1.192 incorporated by reference in 10 CFR 50.55a).	3.9.6.3
COL Item 3.9-7	Not used in NuScale DCA Part 2.	3.9
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

The staff finds COL Items 3.9-4 and 3.9-5 to be acceptable in specifying that a COL applicant that references the NuScale DC must develop a PST and IST program to satisfy 10 CFR 50.55a(f) requirements specific to a COL applicant.

The initial submittal of DCA Part 2, Tier 2, Section 3.9.6.3, included a COL information item stating that a COL applicant that references the NuScale Power Plant DC will identify any site-specific valves and provide inservice testing in accordance with the latest endorsed ASME OM Code with addenda incorporated by reference in 10 CFR 50.55a 18 months prior to the date for initial fuel load.

On September 1, 2017, the staff issued **RAI 8959, Question 03.09.06-27** (ADAMS Accession No. ML17244A097), requesting the applicant to clarify this COL information item to correct the time period prior to initial fuel load consistent with the current regulations and to include the implementation milestones. In its October 11, 2017, response (ADAMS Accession No. ML17284A778) to that RAI, the applicant explained that the COL information item in DCA Part 2, Tier 2, Section 3.9.6.3, would be numbered as COL Item 3.9-6 and would state the following:

> *A COL applicant that references the NuScale design certification will identify any site-specific valves, implementation milestones, and the applicable ASME 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 ASME Code Cases listed in Regulatory Guide 1.192 incorporated by reference in 10 CFR 50.55a).*

In addition, the applicant stated that it would update DCA Part 2, Tier 2, Table 1.8-2, to include the new language of COL Item 3.9-6. The staff confirmed that COL Item 3.9-6 has been updated in DCA Part 2. Therefore, **RAI 8959, Question 03.09.06-27**, is resolved and closed.

In **RAI 8954, Question 03.09.03-12**, the staff requested the applicant to provide a full description of the IST program for POVs for a NuScale Power Plant or to specify a COL action item regarding POV testing. In its response, the applicant proposed a revision to DCA Part 2, Tier 2, Section 3.9.6, to fully describe the IST program for POVs, and also provided proposed COL Items 3.9-8 and 3.9-9 to address test procedures for HOV performance and ECCS valve performance, respectively. The staff determined that the proposed COL Items 3.9-8 and 3.9-9 as updated in DCA Part 2 provide appropriate direction for the COL applicant regarding POV performance and ECCS valve performance.

3.9.6.6 Conclusion

In light of the open item described above, the staff is not able to conclude that the NuScale DCA provides assurance that the IST provisions of the ASME OM Code 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. Following resolution of the open item, the staff will determine whether the descriptions of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints for the NuScale Power Plant design provided by the applicant comply with applicable NRC regulations. The staff will also determine whether the descriptions are acceptable for incorporation by reference in a final safety analysis report submitted by a COL applicant referencing the NuScale design, when supplemented as necessary to provide a full description of those programs, consistent with the Commission policy on operational programs for new reactors.

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, Revision 0, 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, 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, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." 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, "Module Specific Mechanical and Electrical/I&C Equipment," lists the equipment addressed by the ITAAC of Table 2.8-2, "Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria." Section 14.3 of this SER discusses NuScale ITAAC.

Technical Specifications: There are no TS associated with DCA Part 2, Tier 2, Section 3.11

Technical Reports: There are no Technical Reports associated with DCA Part 2, Tier 2, Section 3.11

3.10.3 Regulatory Basis

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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- RG 1.92

- RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants”

- RG 1.100

3.10.4 Technical Evaluation

The staff performed its review of DCA Part 2, Tier 2, Revision 0, 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. 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.” DCA Part 2, Tier 2, Section 3.10.2.1, is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.iii.

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.” DCA Part 2, Tier 2, Section 3.10.1.2, is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.iv.

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.

vi. DCA Part 2, Tier 2, Section 3.10.2.1, states, “The test input motions are applied to two perpendicular horizontal axes or a vertical and a horizontal axis for the seismic and dynamic part of the load unless it can be shown that the sensitivity of the equipment response to vibratory motion in the horizontal direction is insignificant. To avoid an exclusively rectilinear input motion, the time phasing of the inputs in each direction are chosen carefully. Alternatively, the test may be conducted with the horizontal and vertical inputs in-phase and then the test is repeated, after rotating the equipment 90 degrees horizontally with the horizontal and vertical inputs 180 degrees out-of-phase.”

The alternative statement is not in agreement with SRP Section 3.10, which states that an acceptable alternative is to test with vertical and horizontal inputs (SRP Section 3.10, Acceptance Criterion II.1.A.vi) 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. In the SRP, four tests (in-phase/out-phase, rotate 90 degrees, then in-phase/out-phase) are required. However, DCA Part 2, Tier 2, Section 3.10.2.1, shows only two tests (in-phase, rotate 90 degrees, then out-phase). Additionally, SRP Section 3.10 states that, for the seismic and dynamic portion of the loads, 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.

DCA Part 2, Tier 2, Section 3.10.2.1, states, “the test input motions are applied to two perpendicular horizontal axes or a vertical and a horizontal axis for the seismic and dynamic part of the load unless it can be shown that the sensitivity of the equipment response to vibratory motion in the horizontal direction is insignificant.”

In **RAI 8890, Question 03.10-1**, dated September 1, 2017 (ADAMS Accession No. ML17244A855), the staff requested the applicant to justify these deviations or to revise DCA Part 2. In its October 31, 2017, response (ADAMS Accession No. ML17304A918) to that RAI, the applicant provided a markup of DCA Part 2, Tier 2, Section 3.10.2.1, to indicate that the equipment is tested with vertical and horizontal inputs in-phase and out-of-phase and then the equipment is rotated 90 degrees and the test repeated. The markup of DCA Part 2 satisfactorily addresses the testing requirements for equipment and is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.vi. The staff verified that this markup is included in Revision 2 of DCA Part 2. Therefore**, RAI 8890, Question 03.10-1**, is resolved and closed.

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.” DCA Part 2, Tier 2, Section 3.10.2.1, 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.”

DCA Part 2, Tier 2, Section 3.10.3, 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.” DCA Part 2, Tier 2, Section 3.10.2.1, is consistent with SRP Section 3.10, Acceptance Criterion II.1.A.ix.

x. SRP 3.10 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.” 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 3.9.2 of this SER reviews the damping values to be used. 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,” 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, as ASME QME-1-2007 and IEEE Std. 344 will be used, which 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.

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, "ASME BPV Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures." Section 3.9.3 of this report addresses this topic.

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 this criterion is satisfied.

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. In 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 this criterion.

C. verification of seismic and dynamic qualification

SRP Section 3.10.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 4 of SRP Section 3.10.

Based on the above evaluation of SRP criteria i through xiv for the qualification for 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 RG 1.97 instrumentation. 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 RCBP 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 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.

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 5, specifies that the qualification program for valves that are part of the RCBP 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, states 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 CRDMs 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.

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 the DCA Part 2 have 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
- test response spectra
- 5 (OBE)+1SSE or 2 SSE requirement
- the requirement to have seismic testing

The staff found that the design specifications were incomplete and require revision to address the deficiencies indicated in the audit report. NuScale committed to addressing these items, as indicated in the audit report. The staff has closed the audit and considers resolution of the audit items as confirmatory items and will be tracked as **Confirmatory Item 03.10-1**. The staff will confirm resolution of the items identified in the audit summary report in a followup audit, as appropriate.

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 6.A through 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, as 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—

   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. Tier 2, Section 3.10, the applicant does not require in-plant testing.

C. EQRFs contain—

   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 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, Sections 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 (1) 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 (2) 3.10-2	A 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 SSCs that require seismic qualification.	3.10.4
COL Item (3) 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

Subject to completion of the confirmatory actions, 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

Note: Section 3.11 contains issues without a mutually understood and clearly defined path toward resolution related to environmental qualification of electrical equipment. The information needed to complete Section 3.11 evaluation was recently provided as part of the topical report, TR-0915-17565 "Accident Source Term Methodology Revision 3 (ADAMS Accession No.

ML19112A171) which was submitted on April 21, 2019. This TR is currently undergoing an acceptance review. Therefore, Section 3.11 of this safety evaluation will be conducted in parallel with the review of the TR, and a complete Section 3.11 safety evaluation will be documented in the Phase 4 Chapter 3 SE.

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 environmental qualification 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 environmental qualification 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). The staff reviewed that the applicant's environmental qualification program conforms with 10 CFR 50.49 and that the set of equipment to be environmentally qualified includes, as appropriate, safety-related equipment, equipment that is nonsafety-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 environmental qualification 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 environmental qualification 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 environmental qualification of equipment specific to each NPM. Tier 1, Section 3.14, provides environmental qualification 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 environmental qualification 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 environmental qualification 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 environmental qualification 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.

DCA Part 2, Tier 2, Section 3.11 and Appendix 3C, describe the NuScale process for the environmental qualification 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, "Qualification of Active Mechanical Equipment Use in Nuclear Power Plants." 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"; DCA Part 2, Tier 2, Section 3.9.6 and Section 3.10, respectively, address these topics.

ITAAC: ITAAC related to environmental qualification are addressed in SER Section 14.3.6. As**Technical Specifications:** There are no TS for this area of review.

Technical Reports: The following technical report contains relevant information for this review:

- Appendix 3C, "Methodology for Environmental Qualification of Electrical and Mechanical Equipment," Rev. 0.

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.

- 10 CFR Part 50, Appendix A, GDC 1, requires 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.

- 10 CFR Part 50, Appendix A, GDC 2, requires 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.

- 10 CFR Part 50, Appendix A, GDC 4, requires 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.

- 10 CFR Part 50, Appendix A, 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, “Test Control,” requires a test control plan to be established to ensure that all tests needed to demonstrate a component’s capability to perform satisfactorily in service be identified and performed in accordance with written procedures that incorporate the requirements and acceptance limits contained in applicable design documents.

- 10 CFR Part 50, Appendix B, Criterion XVII, “Quality Assurance Records,” requires that sufficient records be maintained to furnish evidence of activities affecting quality.

- 10 CFR 52.47 states, in part, that the Commission will require, before DC, 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 environmental qualification 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, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” (ADAMS Accession No. ML003740271) provides the principal guidance for implementing the requirements and criteria of 10 CFR 50.49 for environmental qualification of electrical equipment that is important to safety and located in a harsh environment.

- NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” (ADAMS Accession No. ML031480402) Category I guidance may be used if relevant guidance is not provided in RG 1.89.

- RG 1.40, "Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants," (ADAMS Accession No. ML093080087) endorses IEEE Std. 334, "IEEE Trial Use Guide for Type Tests of Continuous-Duty Class 1 Motors Installed Inside the Containment of Nuclear Power Generating Stations."

- RG 1.63 (ADAMS Accession No. ML003740219) endorses IEEE Std. 317, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations."

- RG 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants," (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 (ADAMS Accession No. ML061580448) provides guidance acceptable to the staff for the environmental qualification 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, (ADAMS Accession No. ML091320468) endorses, with exceptions and clarification, ASME Standard QME-1-2007 for the qualification of nonmetallic parts of active mechanical equipment.

- RG 1.156, "Qualification of Connection Assemblies for Nuclear Power Plants," (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, "Qualification of Safety-Related Vented Lead-Acid Storage Batteries for Nuclear Power Plants," (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 environmental qualification of Class 1E lead storage batteries and should be used in conjunction with NUREG-0588 and RG 1.89, as appropriate, for evaluating the environmental qualification of lead storage batteries.

- RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio--Frequency Interference in Safety-Related Instrumentation and Control Systems," (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, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (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, “Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants,” (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 environmental qualification 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 equipment 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 (e.g., LOCA and high-energy line break) 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), (b)(2), and (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 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 checked for electrical supporting safety systems that are not safety-related but was unable to clearly identify whether the NuScale design covers electrical equipment that is not safety-related in DCA Part 2, Tier 2, Table 3.11-1. In addition, the staff issued **RAI 8788, Question 08.01-1**, dated May 22, 2017 (ADAMS Accession No. ML17142A389), requesting the applicant to clarify whether electrical penetration assemblies (EPAs) in the Class 1E containment conform to GDC 50. In its July 19, 2017, response (ADAMS Accession No. ML17200D160) to that RAI, NuScale stated that containment EPAs comply with GDC 50. Based on the response, the staff requested the applicant via conference call to revise DCA Part 2, Tier 2, Table 3.11-1, to include EPAs to ensure that the assemblies will be environmentally qualified. In addition, the staff requested the applicant to provide a revised version of DCA Part 2, Tier 2, Table 3.11-1, to include equipment supporting safety systems that are not safety-related, such as cables.

GDC 50 requires, in part, that EPAs be designed to maintain containment integrity during normal and accident conditions. Since they are required to function during and after a DBE to maintain containment integrity, EPAs must be environmentally qualified.

On November 3, 2017, NuScale supplemented the response (ADAMS Accession No. ML17310A296) to **RAI 8788, Question 08.01-1**, based on the staff comments. As part of the supplement, NuScale proposed to revise DCA Part 2, Tier 1, Section 2.8, to include the information related to the EPAs. As part of the revisions to 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. NuScale also proposed to revise DCA Part 2, Tier 2, Table 3.11-1, to clarify the EPA component descriptions for environmental qualification purposes and to add note 6 to address equipment qualification of items such as cables, connectors, electrical splices, conduit seals, thread sealants, terminal blocks, or lubricants through commodities.

The staff has evaluated the supplemental response to **RAI 8788, Question 08.01-1**, and finds acceptable the clarification and revision to clearly identify EPAs in DCA Part 2, Tier 2, Table 3.11-1. However, the staff finds that the equipment listed in note 6 of DCA Part 2, Tier 2, Table 3.11-1, should be incorporated into the equipment qualification list to ensure the completeness of the list, as required in 10 CFR 52.47(a)(13). In addition, the staff noted in the revision of Table 3.11-1 that some of the EPAs are not considered Class 1E; therefore, the staff issued **RAI 9347, Question 3.11-18,** to ask the applicant to incorporate in DCA Part 2, Tier 2, Table 3.11-1, the equipment listed in note 6. The staff also issued **RAI 9038, Question 08.01-2, Part b**, to request that NuScale explain why the I&C equipment, PZR heater power, and CRD power EPAs are not Class 1E in the revisions to DCA Part 2, Tier 1, Table 2.8-1. Therefore, **RAI 8894, Question 03.11-15**, is being tracked as **Open Item 03.11-1** because the applicant needs to address staff concerns about the EPAs classified as non-Class 1E.

The equipment important to safety that is subject to environmental qualification 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. However, the staff found the information in DCA Part 2, Tier 2, Section 3.11.5.1, “Chemical Environments,” insufficient to address chemical effects.

The staff reviewed the information in DCA Part 2, Tier 2, Section 3.11.5.1, 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.*

The staff reviewed the information in DCA Part 2, Tier 2, Appendix 3.C, to verify the chemical environments parameters but was not able to find the information as specified in DCA Part 2, Tier 2, Section 3.11.5.1. The staff issued **RAI 9015, Question 3.11-16**, dated July 30, 2017, (ADAMS Accession No. ML17211A005), to request clarification of where the environmental parameters related to chemical effects are located or incorporation of the information as specified in DCA Part 2.

In its September 11, 2017, response (ADAMS Accession No. ML17254B078) to **RAI 9015, Question 3.11-16**, the applicant proposed to incorporate the information related to chemical effects. Specifically, the applicant revised Appendix 3.C, Table 3C-6, "Normal Operating Environmental Conditions," to incorporate note 4, which 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." In addition, the applicant revised Appendix 3.C, Table 3C-7, "Design Basis Event Environmental Conditions," to incorporate note 4, which 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). The staff finds this response acceptable. Therefore, **RAI 9015, Question 3.11-16**, is now **Confirmatory Item 03.11-1**, pending verification that the proposed changes are incorporated in the next DCA Part 2 revision.

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 these methods are acceptable for environmental qualification 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. Since the EPAs are required to function during and after a DBE to maintain containment integrity, they are environmentally qualified. However, in its response to **RAI 8788, Question 08.01-1**, the applicant revised DCA Part 2, Tier 1, Table 2.8-1, which states that the some EPAs are not Class 1E, including the I&C, pressurizer heater power, and CRD power EPAs. Since EPAs must be designed to maintain containment integrity during normal and accident conditions, the staff issued **RAI 9038, Question 08.01-2, Part b**, requesting NuScale to explain why the I&C, pressurizer heater power, and CRD power EPAs are not Class 1E in the revisions to DCA Part 2, Tier 1, Table 2.8-1. The staff is waiting for the response to this RAI, to verify the completeness of the equipment qualification list. Therefore, **RAI 8894, Question 03.11-15**, is being tracked as **Open Item 03.11-1** because the applicant needs to address the staff concerns about the EPAs classified as non-Class 1E.

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. The staff is waiting for information requested in **RAI 9038, Question 08.01-2, Part b**, to verify the completeness of the equipment qualification list, in order to determine whether 10 CFR 50.49 is met. This is being tracked as **Open Item 03.11-2**.

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 environmental qualification 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 RG 1.89 by following the requirements of 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 principal design criteria for the facility (the GDC in Appendix A to 10 CFR Part 50 establish minimum requirements for the principal design criteria), (2) the design bases and the relation of the design bases to the principal design criteria, 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 equipment qualification 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 follows documentation requirements specified in IEEE Std. 323-1974, as endorsed by RG 1.89, and provides assurance that the equipment qualification 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 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 environmental qualification, 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 environmental qualification of electrical and mechanical equipment. Appendix 3C states that equipment subject to a harsh environment will be environmentally qualified using IEEE Std. 323-1974 and equipment subject to a harsh environment will be qualified using IEEE Std. 323-2003. Therefore, the staff finds that the equipment subject to environmental qualification 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 environmental qualification provides reasonable assurance that the equipment can perform during and after a DBE. Environmental qualification 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.

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 environmental qualification 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 environmental qualification 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 environmental qualification program described in that section includes environmental qualification 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, 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 environmental qualification.

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, 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

DCA Part 2, Tier 2, Section 3.11.6 and Appendix 3C.4, states 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. By letter dated October 17, 2018 (ADAMS Accession No. ML18290A557), NuScale proposed the following exceptions when qualifying nonmetallic parts of mechanical equipment in accordance with Appendix QR-B and provided a proposed markup of the changes to DCA Part 2, Tier 2, Section 3.11.6 and Appendix 3C. These changes are summarized below.

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 Nuclear Quality Assurance (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.

The staff will track the changes to DCA Part 2, Tier 2, Section 3.11.6 and Appendix 3C, that address the above exceptions to Appendix QR-B of ASME QME-1 as **Confirmatory Item 03.11-2** until they are incorporated into a future revision of DCA Part 2.

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 environmental qualification, the EQRF lists the environmental parameters and the qualification process.

The applicant stated 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-1-2007, Appendix QR-B. The applicant provided supplemental information by letter dated August 17, 2017 (ADAMS Accession No. ML17229B488), stating that safety-related mechanical equipment will be qualified in accordance with ASME QME-1-2007, Appendix QR-B, as accepted in RG 1.100, Revision 3. Also, in the same letter, 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 environmental qualification 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 and procurement specifications as discussed in the introduction of 10 CFR 52.47 for the environmental qualification of nonmetallic parts of safety-related mechanical equipment. The staff has closed its review of specific audit items and has identified some of the audit items as confirmatory that will be resolved when NuScale notifies the NRC that those items have been resolved, as identified in the audit summary report (ADAMS

Accession No. ML19018A140). Therefore, the staff will track the resolution of the design specification audit followup items as **Confirmatory Item 03.11-3**.

3.11.4.3   Environmental Qualification to the Radiation Environment

The staff reviewed DCA Part 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, 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.

DCA Part 2, Section 3.11.1.2, specifies that plant areas in which the radiation levels exceed $1x10^4$rads gamma for electrical and mechanical equipment including nonmetallics or consumables (e.g., O-rings, seals, packing, gaskets, lube oil, diaphragms) and 1.0E03 rads gamma for electronic devices and components are in a harsh environment and must be environmentally qualified. 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 when such effects are believed to be significant. In DCA Part 2, Appendix 3C, Table 3C-6 provides the radiation total neutron and gamma doses during normal operation, and Table 3C-8, "Accident EQ Radiation Dose," provides the accident integrated dose.

Normal operational dose includes neutron and neutron-induced gamma radiation from the fission process, gamma radiation from fission products, 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 equipment qualification 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 (see the response to **RAI 8759, Question 12.02-1** (ADAMS Accession No. ML18080A162)). 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. 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.

The accident source term methodology topical report TR-0915-17565, "Accident Source Term Methodology," Revision 2, dated September 11, 2017 (ADAMS Accession No. ML17254B0468) provides information on accident source terms, in combination with the accident source term information in DCA Part 2, Chapters 12 and 15.

While DCA Part 2, Appendix C, Table 3C-6, provides the total integrated dose (TID) inside of containment in zones A through I, it does not provide the radiation dose for radiation zones J through N, which cover other areas outside of containment and inside the RXB that may contain equipment that must be qualified. Therefore, the staff issued **RAI 8837, Question 03.11-1,** requesting that the applicant provide the TID for these areas and describe how it calculated the TID. In the response (ADAMS Accession No. ML18138A383) to **RAI 8837, Question 03.11-1**, the applicant updated Table 3C-6 to provide the normal operation radiation doses for radiation zones J through N and also indicated that it will revise the topical report on accident source term methodology (TR-0915-17565, Revision 3) at a future date and proposed revisions to DCA Part 2, Appendix 3C, Table 3C-8, based on the revised methodology. The staff cannot proceed with the review of the accident source term until it receives the topical report. In the response to **RAI 8837, Question 03.11-1**, the applicant also provided revised normal operation TID information in DCA Part 2, Appendix 3C, Table 3C-6. The applicant revised Table 3C-6 in Supplement 1 of its November 28, 2018, response (ADAMS Accession No. ML18332A397) to RAI 9291, Question 12.02-24, dated January 29, 2018, in which it revised neutron and gamma dose rates based on revised modeling assumptions. The staff is still evaluating this revision, which remains an open item in Chapter 12 of the SER. In addition, in **RAI 9282, Question 03.11-17**, dated January 9, 2018, the staff requested the applicant to provide additional information on the neutron spectrum (see the discussion of **RAI 9282, Question 03.11-17**, in the next paragraph). Finally, in the response (ADAMS Accession No. ML18320A254) to **RAI 9447, Question 03.11-19**, the applicant provided information on a revised bioshield design, which the staff is currently reviewing. As a result of the above, the staff cannot yet reach a conclusion on the normal operation neutron and neutron-induced gamma dose rates provided in Table 3C-6 until it completes the review of these changes. Thus, the normal and accident TID information requested in **RAI 8837, Question 03.11-1**, is being tracked as **Open Item 03.11-3**.

In addition, the applicant did not provide well-defined neutron flux or energy spectrum information. Therefore, the staff issued **RAI 9282, Question 03.11-17**, requesting that the applicant identify and describe the methods, models, and assumptions used to calculate the neutron spectrum and flux above the top of the PZR, inside the containment vessel, to evaluate the potential impact on equipment located in these areas. In the responses dated March 8, 2018, and December 27, 2018 (ADAMS Accession Nos. ML18067A570 and ML18361A905) to **RAI 9282, Question 03.11-17**, NuScale provided information on some modeling changes made, which revised the neutron and neutron-induced gamma dose rates in the response (ADAMS Accession No. ML18332A397) to **RAI 9291, Question 12.02-24**. However, the applicant did not provide the requested neutron flux or energy spectrum information or information adequate for the staff to reach a conclusion. Therefore, the staff is planning an upcoming audit to review information related to neutron flux and energy spectrum. As a result, **RAI 9282, Question 03.11-17**, is being tracked as **Open Item 03.11-4** until the staff completes the audit and determines if the information provided is adequate.

The staff noted that DCA Part 2, Appendix 3C, Table 3C-8, indicates that it includes fission gammas and gammas from neutrons. However, the reactor is expected to be shut down during accident conditions and therefore should not emit a significant quantity of neutrons during shutdown. This led the staff to issue **RAI 8837, Question 03.11-2**, requesting that the applicant explain this apparent discrepancy. In its August 3, 2017, response (ADAMS Accession No. ML17215A546) to that RAI, the applicant specified that Table 3C-8 provides the TID from both normal operating conditions and accident conditions. However, the staff determined that this was inconsistent with Table 3C-6 because some of the TIDs in Table 3C-6 were higher than the values in Table 3C-8. In Supplement 1 of the response, dated October 20, 2017 (ADAMS Accession No. ML17293A170), to **RAI 8837, Question 03.11-2**, the applicant clarified that Table 3C-8 provides only accident doses, which do not include gammas from neutrons and coolant, and proposed to revise DCA Part 2, Appendix 3C, Table 3C-8 accordingly. However, in reviewing the response and the revised Table 3C-8, the staff still did not understand how the values in the table were obtained and was unclear if the maximum hypothetical accident described in DCA Part 2, Chapter 15, was the basis for the values in the table. As previously noted, the applicant indicated that it will provide a revised topical report on the accident source term (TR-0915-17565, Revision 3) at a future date, based on a revised accident source term methodology. The staff cannot proceed with the review of the accident source term until it receives the topical report. Therefore, **RAI 8837, Question 03.11-2**, is being tracked as **Open Item 03.11-5**.

In reviewing Table 3C-8, the staff also noted that the accident integrated doses 1 hour after an accident in all areas were zero. The staff was unclear as to how this conforms to the release timings in RG 1.183, which the applicant references; therefore, the staff issued **RAI 8837, Question 03.11-3**, requesting the applicant to explain how Section 3.3 of RG 1.183 was implemented. In response (ADAMS Accession No. ML18138A383) to that RAI, the applicant provided revisions to DCA Part 2, Tier 2, Appendix 3C, Table 3C-8, and specified that it will provide a revised topical report on accident source term (TR-0915-17565, Revision 3) related to these changes. The staff cannot proceed with the review of the accident source term until it receives the topical report. Therefore, **RAI 8837, Question 03.11-3**, is being tracked as **Open Item 03.11-6**.

The staff also noted that neither Table 3C-8 nor the text of DCA Part 2 describes the assumptions used to determine the radiological conditions outside of containment and inside the plant for EQ purposes during and following accidents. Therefore, the staff issued **RAI 8837, Question 03.11-4**, requesting the applicant to provide this information. In the response (ADAMS Accession No. ML18138A383) to **RAI 8837, Question 03.11-4**, the applicant provided revisions to DCA Part 2, Tier 2, Section 3.11, and specified that it will provide a revised topical report on the accident source term (TR-0915-17565, Revision 3) related to these changes. The staff cannot proceed with the review of the accident source term until it receives the topical report. Therefore, **RAI 8837, Question 03.11-4**, is being tracked as **Open Item 03.11-7**.

In reviewing the application, the staff was unclear as to the scope of accidents or accident conditions considered in the equipment qualification analysis. Therefore, in **RAI 8837, Question 03.11-5**, the staff requested additional information. In response (ADAMS Accession No. ML17293A170), the applicant indicated that the spectrum of small-break LOCAs, spectrum of high- and moderate-energy line breaks, rod ejection accident, and fuel handling accidents were considered. However, the applicant did not specify whether the maximum hypothetical accident described in UDCA Part 2, Chapter 15, was considered. However, as previously noted, the applicant indicated that it will provide a revised topical report on the accident source term (TR-0915-17565, Revision 3) at a future date, based on a revised accident source term methodology. The staff cannot proceed with the review of the accident source term until it receives the topical report. Therefore, the staff is tracking **RAI 8837, Question 03.11-5**, as **Open Item 03.11-8**.

As discussed previously, while DCA Part 2 specifies that synergistic effects are considered, DCA Part 2, Appendix 3C, Section 3C.4.3 states, “In general, dose rate effects occur over long periods, and, therefore, need only be addressed during the radiation conditions that occur during normal operations.” The staff issued **RAI 8837, Question 03.11-**6, requesting that the applicant provide additional justification for excluding synergistic effects following an accident. In the response (ADAMS Accession No. ML17215A546) to **RAI 8837, Question 03.11-6**, the applicant specified that normal operation plus accident dose was considered, but the synergistic effects during an accident need not be considered. During the radiation protection audit, the applicant indicated that the guidance of NUREG/CR-4301, “Status Report on Equipment Qualification Issues Research and Resolution,” issued November 1986, and NUREG/CR-3629, “The Effect of Thermal and Irradiation Aging Simulation Procedures on Polymer Properties,” issued April 1984, provide justification for not considering synergistic effects. The staff did not find this guidance to provide appropriate justification, because they indicate that for certain materials and conditions consideration of synergistic effects (mainly thermal and radiation) are important. However, as previously noted, the applicant indicated that it will provide a revised topical report on the accident source term (TR-0915-17565, Revision 3) at a future date, based on a revised accident source term methodology. The staff cannot proceed with the review of the accident source term until it receives the topical report. Therefore, **RAI 8837, Question 03.11-6,** is being tracked as **Open Item 03.11-9**.

DCA Part 2, Appendix 3C, Section 3C.4.4, states, "Equipment required to be environmentally qualified has one or more of the following design functions related to safety: reactor trip, engineered safeguards actuation, post-accident monitoring, or containment isolation." It is not clear to the staff how this list was developed. In 10 CFR 50.49, the NRC specifies that equipment required to maintain safe shutdown and equipment that is not safety-related whose failure could prevent satisfactory performance of a safety function, must be qualified. Therefore, the staff issued **RAI 8837, Question 03.11-7**, requesting the applicant to explain how this list comprehensively meets the requirements of 10 CFR 50.49, to update DCA Part 2 accordingly, and to ensure that equipment does not need to be added to DCA Part 2, Table 3.11-1, as a result. In the response to **RAI 8837, Question 03.11-7**, the applicant clarified that safety-related equipment is included within the EQ program and that the potential for failure of electrical equipment that is not safety-related to impact or prevent the accomplishment of a function covered under 10 CFR 50.49b(1) or b(3) has been considered. Because of the passive nature of the NPM, electrical power is not needed to mitigate DBAs. The lack of a safety-related electrical power system eliminates many of the electrical interactions related to failure of electrical equipment that is not safety-related causing a loss of electrical power to an electrical component that is performing a safety-related function during or following a DBA. The design of the highly reliable DC power system (EDSS) and normal DC power system (EDN) electrical systems does provide electrical separation between the EDSS, which provides electrical power to equipment subject to EQ requirements. The electrical isolation between the EDSS and EDN has been credited in preventing an electrical fault in the EDN from affecting the equipment powered by EDSS. As a result, currently, no 10 CFR 50.49b(2) items require qualification to harsh environmental conditions. Finally, the applicant also specified that information associated with RG 1.97, Revision 4, Category B, C, and D variables has been included in the EQ program (there are no Category A variables in the NuScale design). The staff finds this consistent with 10 CFR 50.49 and GDC 4 and acceptable. Therefore, **RAI 8837, Question 03.11-7**, is tracked as **Confirmatory Item 03.11-4**.

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." This description does not explain how equipment located in harsh conditions will be monitored and managed throughout plant life. As discussed often 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 possibly result in a TID (including consideration for postulated accidents) that exceeds the TID that the equipment in that area was designed to withstand. Therefore, the staff issued **RAI 8837, Question 03.11-8**, requesting the applicant to discuss in DCA Part 2 any actions that may be necessary if the measured dose is higher than the calculated dose for equipment located in harsh environments.

In the response (ADAMS Accession No. ML17215A546) to **RAI 8837, Question 03.11-8**, the applicant proposed adding COL Item 3.11-4 to DCA Part 2. This COL item states the following:

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. Therefore, the response is acceptable and will be tracked as **Confirmatory Item 03.11-5**.

In its review of Section 3.11 and Appendix 3C, the staff found that it was unclear whether the EQ analysis considered CVCS line breaks. Therefore, the staff issued **RAI 8837, Question 03.11-9**. In the response (ADAMS Accession No. ML17215A546) to **RAI 8837, Question 03.11-9**, the applicant specified that the EQ analysis evaluated all DBAs. Specifically, CVCS line breaks outside of containment would not affect the TID of equipment outside of containment. The analysis accounts for the dose from the source in an intact CVCS line. However, the applicant indicated that if the line break occurs outside of the module bay, the CVCS source (contaminated water) would fall down the pipe chase and collect in the sumps at the bottom level of the plant where no equipment is being qualified. The applicant indicated that (1) since the CVCS line is isolated in an accident, this would drain the CVCS line near equipment and actually lower the dose to that equipment, (2) any additional airborne dose from a CVCS line break in the module bay is bounded by other accidents, such as the fuel handling accident, for which the doses were added for EQ analysis, and (3) the potential airborne dose from a CVCS line break outside of the module bay would be limited in consequence and manifest in the lower regions of the RXB, below the pipe chase where the coolant collects in the drains and sumps, which is not near equipment needing qualification.

However, the staff determined that primary coolant liquid samples are taken through the CVCS system, and opening the CVCS CIVs allows flow through the CVCS lines and sample lines. Similarly, the CES and containment flooding and drain system are used to take gaseous samples. Therefore, in the process of taking postaccident samples, in accordance with 10 CFR 50.34(f)(2)(viii), these systems would be recirculating fluid from the reactor core which could potentially be highly contaminated if fuel damage had occurred. In 10 CFR 50.49(e)(4), the NRC requires that the radiation from recirculating fluids must be considered in the EQ program for equipment located near the recirculating lines. RG 1.183, Appendix I, Section 10, indicates that the recirculating fluid sources should include containment sump water outside of containment, including sample systems. Section 10 of Appendix I to RG 1.183 also specifies that the EQ analysis should consider radiation from airborne activity, including leakage from recirculating systems. However, as previously noted, the applicant has indicated that it will provide a revised topical report on the accident source term (TR-0915 17565, Revision 3) at a future date, based on a revised accident source term methodology. In addition, the applicant has indicated that it may request an exemption from the requirements of 10 CFR 50.34(f)(2)(viii). The staff cannot proceed with the review of this item until it has received the accident source term topical report and, potentially, the exemption request from the requirements of 10 CFR 50.34(f)(2)(viii). Therefore, **RAI 8837, Question 03.11-9**, is being tracked as **Open Item 03.11-10**.

The staff noted that DCA Part 2, Appendix 3C, Section 3C.5, under Environmental Qualification of Mechanical Equipment, states that "Equipment that only has the design function related to safety of maintaining its structural integrity, for support or to protect the integrity of a pressure boundary, is qualified in accordance with the requirements specified in Section 5.2.1." There is no discussion of the effects of radiation on equipment in DCA Part 2, Section 5.2.1. Therefore, the staff issued **RAI 8837, Question 03.11-10**, requesting the applicant to provide this information. In the response (ADAMS Accession No. ML17215A546) to **RAI 8837, Question 03.11-10**, the applicant proposed to update DCA Part 2, Appendix 3C, Section 3C.5, to correct the statement indicating that equipment is qualified in accordance with the requirements specified in DCA Part 2, Section 5.2.1, to state that the equipment is qualified in accordance with DCA Part 2, Section 3.11. The staff evaluated the applicant's proposal and found the update to be accurate and acceptable. Therefore, **RAI 8837, Question 03.11-10**, will be tracked as **Confirmatory Item 03.11-6**.

DCA Part 2, Appendix 3C, Section 3C.4.2, states that "Radiation aging may be performed separately from the accident radiation exposure or the accident radiation exposure may be performed as part of the radiation aging." The intent of this statement is unclear. Therefore, the staff issued **RAI 8837, Question 03.11-11**, requesting the applicant to explain the intent of the statement.

In the response (ADAMS Accession No. ML17215A546) to **RAI 8837, Question 03.11-11**, the applicant clarified that radiation aging during a qualification test program may be done separately from or concurrently with the accident radiation exposure. The statement in Section 3C.4.2 indicates that the radiation exposure 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). It is intended to allow 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. It does not mean that the degradation from the accident dose does not need to be considered in combination with the degradation from the normal dose exposure.

The staff found the response to be acceptable because both the normal operation and accident dose are to be considered in total, which is consistent with the requirements of 10 CFR 50.49 and GDC 4. Therefore, the response to **RAI 8837, Question 03.11-11**, is closed.

The staff notes that Chapter 12 of this SER also discusses RAIs that relate to source terms and radiation protection design features associated with the EQ analysis. For example, RAI 9282 requests more information on the neutron spectrum and flux above the top of the PZR, inside the containment vessel. See Chapter 12 of this SER for additional information related to source terms and radiation protection design features used to determine the dose rates for the various areas of the plant.

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 environmental qualification operational program that incorporates the aspects specific to the environmental qualification of mechanical and electrical equipment.	3.11.7

In a letter dated August 17, 2017 (ADAMS Accession No. ML17229B488), the applicant stated that it would revise DCA Part 2, Section 3.11.7, to add aspects specific to the environmental qualification operational program and add COL Item 3.11-3 to specify that a COL applicant that references the NuScale Power Plant DC will implement an environmental qualification operational program that incorporates the aspects specific to the environmental qualification of mechanical and electrical equipment. The staff confirmed that NuScale revised DCA Part 2, Tier 2, as described and in an acceptable manner.

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 EQ of mechanical and electrical equipment. The staff evaluated the information for compliance with the following:

- 10 CFR 52.47(a)(13), which requires the application for a standard DC to include a list of electrical equipment important to safety as required by 10 CFR 50.49(d))
- 10 CFR 52.47, which requires certain procurement specifications to be available for audit
- applicable QA criteria in 10 CFR Part 50, Appendix B
- 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 23, and 10 CFR 50.49
- conformance with applicable RGs and standards committed to by the applicant

The staff also reviewed the COL information items in DCA Part 2, Tier 2, Table 1.8-2.

Based on its review, the staff concludes that the SER remains incomplete pending satisfactory resolution of the **open items** identified and the submittal and review of the topical report on accident source term methodology (TR-0915-17565, Revision 3). The staff will update

Section 3.11 of this SER to reflect the final disposition of the DCA Part 2 application. Additionally, the staff will verify that updates to DCA Part 2 address the **confirmatory items**.

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 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 2, 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 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 NRC Commission's direction in SECY-90-377 (ADAMS Accession No. ML0037078890),

“Requirements for Design Certification under 10 CFR Part 52,” dated November 8, 1990. 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 Code Class 1, 2, and 3 piping systems to comply with ASME Code, Section III, requirements through the completion of ASME Code, Section III, design reports for the ASME Code Class 1, 2, and 3 as-built piping systems. Section 14.3 of this SER discusses NuScale ITAAC.

Technical Specifications: There are no TS associated with DCA Part 2, Tier 2, Section 3.12.

Technical Reports: The TR for this area of review is TR-0916-51502-P, Revision 1, NuScale Power Module Seismic Analysis, issued September 2018 (ADAMS Accession No. ML18271A193).

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 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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

- 10 CFR Part 50, Appendix A, 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 BPV 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 (ADAMS Accession No. ML0037078890)

- SECY-93-087, (ADAMS Accession No. ML003708021)

- SRP Section 3.7.2, “Seismic System Analysis,” Revision 3 (ADAMS Accession No. ML052070318)

- SRP Section 3.7.3, “Seismic Subsystem Analysis,” Revision 4 (ADAMS Accession No. ML13198A239)

- SRP Section 3.9.1, “Special Topics for Mechanical Components,” Revision 3 (ADAMS Accession No. ML070430402)

- SRP Section 3.9.2, “Dynamic Testing and Analysis of Systems, Structures, and Components,” Revision 3 (ADAMS Accession No. ML070230008)

- SRP Section 3.9.3, “ASME BPV Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures,” Revision 3 (ADAMS Accession No. ML14043A231)

- NRC white paper, “Piping Level of Detail for Design Certification” (ADAMS Accession No. ML14065A067)

- NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” issued September 2004 (ADAMS Accession No. ML04357033)

- 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 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. For detailed final ASME Class 1 piping design analysis, it has chosen the CVCS RCS discharge piping. This line 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. Hence, the applicant stated that this line is more challenging for structural piping analysis. For detailed final ASME Class 2 piping analysis, it has chosen the FW line, which is also recommended by the staff's white paper, and therefore is an acceptable choice. DCA Part 2, Tier 2, Section 3.12.1, also identifies preliminary piping analyses performed for DC that include Class 1 lines of the CVCS RCS injection piping, RPV high-point degasification piping and RCS PZR spray; and Class 2 lines of the DHRS and MS up to the first six-way rigid restraint beyond the CIVs. The staff finds that the selection of piping systems shown in DCA Part 2, Tier 2, Section 3.12.1, that are included in the graded approach is based on the safety function, piping size, and layout and that 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 with the exception of the MS, which the paper includes in the detailed piping analyses. Therefore, on August 12, 2017, the staff issued **RAI 9069, Question 3.12-5** (ADAMS Accession No. ML17224A024), requesting the justification for excluding a detailed piping analysis for the MS line.

In DCA Part 2, Tier 2, Section 3.12.1, the applicant showed that in the preliminary and detailed pipe stress evaluations, the only loads considered are those resulting from deadweight, seismic, and thermal expansion. In **RAI 9069, Question 3.12-5** (ADAMS Accession No. ML17224A024), the staff requested the applicant to provide a technical justification for considering, in lieu of all applicable loads, only these loads for piping analysis listed in DCA Part 2, Tier 2, Section 3.12.5.3, "Loadings and Load Combinations."

In DCA Part 2, Tier 2, Section 3.12.1, the applicant also indicated that ASME Class 2 rules may be used for ASME Class 1 piping stress analysis. In **RAI 9069, Question 3.12-5**, the staff requested the applicant to provide its technical justification for the rationale of using ASME Class 2 rules for ASME Class 1 piping.

In its responses dated December 3, 2018, and January 25, 2019 (ADAMS Accession Nos. ML18337A333 and ML19025A290) to **RAI 9069, Question 3.12-5**, the applicant showed that the main steam line inside containment and outside containment to the first six-way anchor restraint beyond the reactor bay wall will be included in the scope for detailed final as-designed piping analyses. It also showed that the preliminary and detailed piping analyses will use all applicable loads mentioned in DCA Part 2, Tier 2, Section 3.12.5.3. In addition, the applicant stated that preliminary stress evaluations of the ASME Class 1 lines will use Class 1 rules. DCA markup pages provided with the response reflect the applicant's response. The staff finds the applicant's response acceptable because, as discussed above, the applicant has complied with the staff's requests in the RAI to follow the guidance found in the NRC white paper "Piping Level of Detail for Design Certification” (ADAMS Accession No. ML14065A067) and SECY-90-377 (ADAMS Accession No. ML003707889, “Requirements for Design Certification under 10 CFR Part 52,” dated November 8, 1990. **RAI 9069, Question 3.12-5**, is being tracked as **Confirmatory Item 03.12-1** for DCA changes.

For the DC, along with documenting the methodology to be used, the applicant should summarize the results of piping stress analyses performed for DC, with full piping analysis reports available for potential auditing. According to the white paper, this information is necessary to enable the staff to make a safety determination of piping issues at the DC stage. The applicant in letters No. RAIO-1218-63709, dated December 5, 2018 (ADAMS Accession No. ML18339A033), and No. RAIO-0119-64288, dated January 24, 2019 (ADAMS Accession No. ML19024A422), 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 meet allowable stress limits in ASME 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 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 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 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 Boiler and Pressure Vessel 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 Code, Section III, 2013 Edition (no addenda). In using ASME Code, Section III, the applicant stated that it follows the regulatory conditions found in 10 CFR 50.55a(b)(1). Hence, the application of ASME 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 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 Code, Section III, requirements through the completion of ASME 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 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 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 that RG 1.84 endorses ASME Code sections and Code Cases that may be applied to ASME Class 1, 2, and 3 piping systems.

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 technique. Piping attached to the NPM uses the integral shield restraints (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, "Seismic Design," 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 envelopes all the individual response spectra for these locations and thus defines a uniform response spectrum (URS). SRP Section 3.7.3, 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

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 square root of the sum of the squares (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 1, 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, Subsection 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 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 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 its December 10, 2018, supplemental response to **RAI 9362, Question 03.08.02-15** (ADAMS Accession No. ML18344A512), the applicant clarified that socket welded joints are not used on lines greater than or equal to 3/4-inch NPS and that any socket welds used on piping less than 3/4-inch NPS conform to 10 CFR 50.55a(b)(1)(ii). Accordingly, the applicant provided markup pages for 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." The staff finds the above statement in DCA Part 2, Tier 2, Section 3.12, and the **RAI 9362, Question 03.08.02-15**, supplemental response markup pages 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. **RAI 9362, Question 03.08.02-15**, is being tracked as **Confirmatory Item 03.08.02-1** for the DCA changes on socket welded connections. SER Section 3.8.2 discusses the disposition of the remainder of **RAI 9362, Question 03.08.02-15**.

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 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 the computer programs to be used in the design of NuScale piping. Piping stress analysis computer programs include ANSYS and AUTOPIPE have been used to validate the ANSYS and AUTOPIPE computer programs. 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.

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. SER Section 3.9.1 presents further evaluation of the acceptance of these programs.

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 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 Code, Section III, methodology and equations, which include evaluations for service levels A, B, C, and D, as well as testing. 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 1/3 of the SSE, the OBE is not considered in the design and showed that the OBE is associated with plant shutdown and 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 high-energy line breaks 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 Code, Section III. The staff concluded that the applicant’s position complies with the requirements of ASME 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 shows 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.

From IEEE 344, Annex D:

(PCE_{max}) = (No. of fractional cycles) × (Percentage of maximum peak cycles)$^{2.5}$

Given data: 20 full SSE vibratory cycles at 1/3 the maximum SSE amplitude, which translates to equivalent maximum peak cycles, $(PCE_{max}) = 20$; and

(Percentage of maximum peak cycles) = 1/3.

By substituting the given data into the above equation and solving for number of fractional cycles, n:

$n = 20/(1/3)^{2.5} = 311.8$ or 312 fractional amplitude SSE cycles.

The staff reviewed the applicant's position on use of seismic cycles for Class 1 piping fatigue evaluation, as shown above, and finds it acceptable as it is consistent with the staff's guidance in SRP Section 3.7.3, Revision 4.

In 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, Revision 1. 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 requirement. 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 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, "Thermal Stresses in Piping Connected to Reactor Cooling Systems," dated June 22, 1988, 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), 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 RCL. 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 1-inch NPS. 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, “Cracking in Feedwater System Piping,” dated October 16, 1979, discusses the effects of thermal stratification in operating reactors in FW lines, and NRC BL-88-11, “Pressurizer Surge Line Thermal Stratification,” dated December 20, 1988, 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 containment vessel 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 with stagnant water at a lower temperature than the FW and, therefore, could create the potential for thermal stratification. Hence, the NRC staff in **RAI 9073, Question 3.12-7** (ML17224A027) requested the applicant to investigate for TASCS susceptibility in the DHRS piping and provide its position for the staff to review.

The applicant in its **RAI 9073, Question 3.12-7** response (ML18038B623) confirmed the DHRS is not isolable from the FW line. As shown in DCA Part 2, Tier 2, Table 2.1-1, the DHRS line is ASME class 2. The applicant noted that the DHRS line is not connected to the reactor coolant system and, therefore, Bulletin 88-08 is not directly applicable to DHRS. The applicant though evaluated the DHRS line for Bulletin 88-08 TASCS effects. The applicant's response identified that performed computational fluid dynamics analysis 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 containment vessel. The staff accepts the applicant's response because it provides reasonable assurance by evaluation that although the DHRS condensate line is susceptible TASCS, fatigue damage is prevented. In addition, the applicant in its response shows that it has incorporated the operating experience from Bulletin 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. Therefore, RAI 179-9073, Question 3.12-7 is resolved and closed.

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.

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 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 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. The ASME BPV Code B31.1, Nonmandatory Appendix II, takes a similar approach to calculating the reaction force resulting from discharge thrust. 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 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 adequate fluid flow path 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 Code were high enough that the functional capability of piping subject to such stresses was questioned. The staff observes that where the ASME 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 Code are considered sufficient to ensure piping functional capability consistent with NUREG-1367. Therefore, the applicant’s use of the ASME 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 seismic anchor motion 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 seismic anchor motion 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 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 Code, Section III, Nonmandatory Appendix Y.

Although the nonmandatory appendices to ASME 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 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 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

DCA Part 2, Tier 2, Section 3.12.3.5, “Damping Values,” and Section 3.12.3.2.2, contains the applicant’s discussion and position on modal damping for composite structures. The staff’s review of this material appears in Section 3.12.4.3.4 of this report.

3.12.4.4.17 Minimum Temperature for Thermal Analyses

According to SRP Section 3.12, Acceptance Criterion II.C.xvii, the stress-free reference temperature for a piping system is defined as a temperature of 70 degrees F. For piping systems that operate at temperatures above 70 degrees F, a thermal expansion analysis should be performed in accordance with ASME 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 150 degrees F or less. The applicant also states 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 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 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 thermal expansion effects on 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 over pressurization 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 rapid 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 Code, Section III, Subsection NF. In addition, for Service Level D, the NuScale pipe support design uses the stress limits in the ASME Code, Section III, Non-Mandatory 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 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 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, "Specification for Safety-Related Steel Structures for Nuclear Facilities." 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 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 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, Revision 2, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," dated November 8, 1979. 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 1/16 inch for SSE loadings and a maximum of 1/8 inch 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 seismic anchor motions 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 3.12.6.8, 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 Code, Section III, Subsection NF. 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, “Applicable Codes,” because ASME BPV Code, Section III, Subsection NF is recommended by SRP Section 3.12 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 non-seismic 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 coefficient of friction.

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 coefficient of friction of 0.3. Its reference for this value is WRC BL 353. The staff notes that the 0.3 coefficient of friction 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 March 2011 (ADAMS Accession No. ML14099A519)). Therefore, the staff accepts the coefficient of friction 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 1/16 inch 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 1/8-inch 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 additional tolerances are acceptable, because they increase 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.vi.

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 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 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 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

Upon the successful resolution of confirmatory items identified above, 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 October 30, 2018, 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). The applicant supplemented this information with letters dated as follows:

- 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 PSI and ISI requirements.

3.13.2 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, 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 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 with the acceptance criteria in SRP Section 3.13.

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 5.4.1.5, “Steam Generator Materials,” discusses the pressure-retaining components that are part of the SGs, including bolting material, which are listed in 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.

Materials Selection

DCA Part 2, Tier 2, Section 3.13.2, “Inservice Inspection Requirements,” 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 a copy of 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 bolting 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, "Materials Selection." This is in accordance with Section II of the ASME BPV Code but has a more restrictive solution temperature range of 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.

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, and 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.

Preservice Inspection Requirements

The applicant stated that the PSI requirements are in accordance with the ASME BPV Code, Section XI.

Certified Material Test Reports (QA Records)

The applicant stated that all 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 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.”

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 describes the inspection for the RPV and CNV main flange bolts (greater than 2 inches in diameter) and pressure-retaining bolting that is 2 inches or less in diameter. The only threaded fasteners greater than 2 inches in diameter are the RPV and CNV main flanges.

DCA Part 2, Tier 2, Table 5.2-6, “Reactor Pressure Vessel Inspection Elements,” and Table 6.2-3, “Containment Vessel Inspection Elements,” states that the threaded fastener Vremoved.

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 NuScale ITAAC.

Technical Specifications: DCA Part 2, Tier 2, Chapter 16, “Technical Specifications,” does not contain TS related to DCA Part 2, Tier 2, Section 6.1.1.

Technical Reports: The staff reviewed the following TRs, which are incorporated by reference in DCA Part 2, Tier 2, Table 1.6-2, “NuScale Referenced Technical Reports”:

- TR-1116-51962-NP, Revision 0, “NuScale Containment Leakage Integrity Assurance Technical Report,” issued December 2016

- TR-0917-56119-P, Revision 0, “CNV Ultimate Pressure Integrity,” issued December 2017

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 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 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 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 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, “Threaded Fasteners—ASME BPV Code Class 1, 2, and 3,” Revision 0, 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 and replaced with RG 1.28 after the most recent version of SRP Section 3.13 was issued in March 2007. In the future, SRP Section 3.13 will be updated to reflect the new staff guidance.

3.13.4 Technical Evaluation

In 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 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 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	RPV21-22	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
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. 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.

The materials selected for use for the threaded fasteners are Alloy 718 and 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 and 10 CFR 50.55a. Since some of the applications are for the RCPB, the staff also finds that GDC 14 and GDC 30 are met.

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. 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 6 inches, 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 operating environments. Furthermore, NuScale proposed water chemistry controls for the primary, secondary, and UHS. DCA Part 2, Tier 2, Section 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 facture 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. However, the staff did review whether the positions in the RG should be considered and found that, since the design is the same, the positions in RG 1.65 apply to the CNV main flange threaded fasteners in the same way and therefore are still 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. In addition, these reports cite other EPRI reports referenced in the SRP and NUREG-1339. 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. 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 to ensure cladding integrity. Since these are augmented inspections for unique portions of the NuScale design, which go beyond the ASME BPV Code requirements, 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 part of the staff's review in Section 3.6.2 of the 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 section of the ASME BPV Code, the staff finds that the applicant meets the requirements of GDC 1 and 10 CFR 50.55a related to fabrication 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 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 these requirements acceptable because the ASME BPV Code, Section III, imposes additional requirements over and above those included in the ASTM materials specifications.

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 are augmented inspections 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.

The RVV and RRV flange connection threaded fasteners are less than 2 inches 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, which go beyond the ASME BPV Code requirements, acceptable as part of the staff's review in Section 3.6.2 of the SER. However, NuScale's RAI response letter dated September 13, 2018 (ADAMS Accession No. ML18256A300) related to the RVV and RRV flange connection threaded fasteners cites the same technical response as another letter, dated December 18, 2017 (ADAMS Accession No. ML17352B263). The staff did not make its finding based on the technical assertions in these letters and does not endorse these assertions.

Compliance with the requirements of the ASME BPV Code, Section XI, also satisfies the regulatory requirements of 10 CFR 50.55a. DCA Part 2, Tier 2, Section 3.13.1.4, Section 3.13.2, Table 3.13-1, and Table 3.13-2, states 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 must 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 reviewed the proposed ITAAC in Section 14.3 of the SER.

3.13.4.8 Technical Specifications

There are no TS 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. Therefore, the staff finds this acceptable in accordance with 10 CFR 50.36, "Technical Specifications."

3.13.4.9 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 in-service inspection program for 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 items 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.

3.13.5 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.