3 DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS AND EQUIPMENT

3.7 Seismic Design

3.7.1 Seismic Design Parameters

3.7.1.1 Introduction

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

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

3.7.1.2 Summary of Application

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

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

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

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

3.7.1.3 Regulatory Basis

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

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

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

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

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

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

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

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

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

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

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

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

3.7.1.4 Technical Evaluation

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

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

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

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

3.7.1.4.1 Design Ground Motion

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

3.7.1.4.2 Certified Seismic Design Response Spectra

FSAR Section 3.7.1.1.1, "Design Ground Motion Response Spectra," applies the design response spectra, which would become the CSDRS once the NuScale FSAR is approved, as an outcrop motion at the finished grade in the free field at the foundation level of the Seismic Category I and II structures. The CSDRS is applied at three mutually orthogonal directions—two horizontal and one vertical. In FSAR Figure 3.7.1-1, "NuScale Horizontal CSDRS at 5

Percent Damping," and Figure 3.7.1-2, "NuScale Vertical CSDRS at 5 Percent Damping," compare the CSDRS and the RG 1.60 spectra at 5-percent damping for the horizontal and vertical directions, respectively. The CSDRS are the same in the two horizontal directions, which are identified as north-south (N-S) and east-west (E-W). The horizontal and vertical components of the CSDRS have a peak ground acceleration of 0.5g and 0.4g, respectively.

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

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

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

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

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

3.7.1.4.3 Certified Seismic Design Response Spectra-High Frequency

FSAR Section 3.7.1.1.1.2, "Certified Seismic Design Response Spectra-High Frequency," describes the CSDRS-HF to include hard rock sites that may also be used for the NuScale design of Seismic Category I structures. The CSDRS-HF has a narrow frequency range below approximately 10 Hz and greater frequency range above approximately 10 Hz than the CSDRS. The CSDRS-HF is applied at three mutually orthogonal directions—two horizontal and one vertical. In FSAR Figure 3.7.1-3, "NuScale Horizontal CSDRS-HF at 5 Percent Damping," and Figure 3.7.1-4, "NuScale Vertical CSDRS-HF at 5 Percent Damping," the applicant compared the CSDRS and the CSDRS-HF at 5-percent damping for the horizontal and vertical directions, respectively. The CSDRS-HF are the same in the two horizontal directions (N-S and E-W). FSAR Table 3.7.1-2, "Certified Seismic Design Response Spectra—High Frequency Control Points at 5 Percent Damping," provides the control points for the CSDRS-HF at 5-percent damping. The peak ground acceleration of the CSDRS-HF is 0.5g for both the horizontal and vertical directions.

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

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

3.7.1.4.4 Design Ground Motion Time Histories

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

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

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

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

<u>Evaluation of CSDRS and CSDRS-HF Compatible Ground Motion Time Histories</u>

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

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

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

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

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

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

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

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

3.7.1.4.5 Percentage of Critical Damping Values

FSAR Section 3.7.1.2, “Percentage of Critical Damping Values,” states that the damping values used for the analysis of the Seismic Category I and II SSCs are based on RG 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants." The staff confirmed that the applicant used values of critical damping that are consistent with those in RG 1.61. The staff finds this acceptable for use in subsequent dynamic analysis.

Structural Damping

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

Soil Damping

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

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

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

3.7.1.4.6 Supporting Media for Seismic Category I Structures

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

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

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

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

3.7.1.5 Combined License Information Items

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

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

Item No.	Description	FSAR Section
COL Item 3.7-1	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific safe shutdown earthquake.	3.7.1

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

3.7.1.6 Conclusion

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

3.7.2 Seismic System Analysis

3.7.2.1 Introduction

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

3.7.2.2 Summary of Application

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

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

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

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

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

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

Technical Reports:

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

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

- NuScale Technical Report, TR-121515-P, Revision 1, “NuScale Power Module Seismic Analysis” (ML24327A037 (proprietary); ML24327A036 (non-proprietary))

3.7.2.3 Regulatory Basis

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

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

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

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

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

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

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

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

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

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

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

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

3.7.2.4 Technical Evaluation

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

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

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

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

3.7.2.4.1 Seismic Analysis Methods

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

Frequency-Domain Soil-Structure-Fluid Interaction Analysis

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

Background

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

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

Analysis Method

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

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

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

Applicability of the Topical Report Methodology to US460 Standard Design

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

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

Computer Programs

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

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

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

3.7.2.4.2 Natural Frequencies and Responses

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

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

3.7.2.4.3 Procedures Used for Analytic Modeling

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

<u>Reactor Building Model</u>

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

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

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

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

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

<u>NuScale Power Module Model</u>

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

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

To validate the simplified NPM model included in the RXB ANSYS model, the applicant performed dynamic modal analysis and discussed the results from the simplified NPM model in comparison with those from the detailed NPM model in Section 4.2 of the NuScale Technical Report, TR-121515-P, Revision 1, "NuScale Power Module Seismic Analysis." The staff noted that the comparison shows closely matching frequencies and mass participation ratios from the two models. Therefore, the staff concluded that the simplified NPM model captures the overall dynamic behavior of the detailed NPM model and therefore is adequate to be included in the RXB ANSYS model for the seismic response analysis of the RXB.

Control Building Model

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

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

Double Building Model

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

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

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

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

Conclusion on Analysis Models

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

Procedures for structural analysis of the RXB and CRB

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

The staff reviewed the applicant’s approach for developing the seismic demand and combining them with other loads to establish the structural design loads for the RXB and CRB and found them acceptable because (1) it uses the seismic design parameters evaluated in SER

Section 3.7.1.4, (2) it uses the finite element models evaluated in SER Section 3.7.2.4.3.1, and (3) the seismic demand is combined with loads from other sources to develop the structural design loads for the buildings in accordance with the guidance in DSRS Section 3.8.4. More detailed staff review on related issues is provided in SER Section 3.8.4.

3.7.2.4.4 Soil-Structure Interaction Analysis

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

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

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

<u>Sensitivity Studies on Parameter Variations</u>

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

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

<u>Effect of Empty Dry Dock</u>

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

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

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

<u>Effect of NPM Modularity</u>

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

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

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

<u>Effect of Potential Soil Separation</u>

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

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

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

Conclusion on Sensitivity Studies

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

3.7.2.4.5 Development of In-Structure Response Spectra

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

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

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

The staff finds the applicant's process for development of the ISRS from response time histories, combining the three directional response time histories at each location using the algebraic sum, computation of the ISRS at a minimum number of frequencies, and the 15-

percent broadening of the peaks in the ISRS, conforms to the guidance in RG 1.122 and meet the acceptance criteria in DSRS Section 3.7.2.II.5 and, therefore, are acceptable.

3.7.2.4.6 Three Components of Earthquake Motion

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

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

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

3.7.2.4.7 Combination of Modal Responses

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

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

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

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

In FSAR Section 3.7.2.8, "Interaction of Non-Seismic Category I Structures with Seismic Category I Structures," the applicant described that the nearby non-Seismic Category I structures are evaluated and concluded that there is no potential for adverse interaction with Seismic Category I SSCs during the SSE conditions. The staff notes that the nearby non-

Seismic Category I structures that are adjacent to the Seismic Category I portions of the RXB and CRB include (1) the RWB that is adjacent to the RXB, and (2) the non-Seismic Category I portion of the CRB that is directly to the north of the Seismic Category I portion of the CRB. The staff also notes that the FSAR includes COL Item 3.7-7 which ensures that an applicant referencing the US460 standard design will confirm that nearby structures exposed to the site-specific SSE will not collapse and adversely affect Seismic Category I portions of the RXB and CRB.

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

3.7.2.4.9 Effects of Parameter Variations on Floor Response Spectra

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

3.7.2.4.10 Use of Constant Vertical Static Factors

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

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

3.7.2.4.11 Method Used to Account for Torsional Effects

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

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

3.7.2.4.12 Comparison of Responses

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

3.7.2.4.13 Analysis Procedure for Damping

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

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

3.7.2.4.14 Determination of Dynamic Stability of Seismic Category I Structures

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

3.7.2.5 Combined License Information Items

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

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

Item No.	Description	FSAR Section
COL Item 3.7-5	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific analysis that assesses the effects of soil separation. The applicant will confirm that the in-structure response spectra in the soil separation cases are bounded by the in-structure response spectra described in Section 3.7.2.	3.7.2
COL Item 3.7-6	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific	3.7.2

	analysis that assesses the effects of non-vertically propagating seismic waves on the free-field ground motions and seismic responses of seismic Category I structures, systems, and components.	
COL Item 3.7-7	An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to a site-specific safe shutdown earthquake will not collapse and adversely affect seismic Category I portions of the Reactor Building and Control Building.	3.7.2
COL Item 3.7-8	An applicant that references the NuScale Power Plant US460 standard design will demonstrate that the site-specific seismic demand is bounded by the Final Safety Analysis Report capacity for an empty dry dock condition.	3.7.2
COL Item 3.7-9	An applicant that references the NuScale Power Plant US460 standard design will perform a soil-structure interaction analysis of the Reactor Building and the Control Building using the NuScale ANSYS models for those structures. The applicant will confirm that the site-specific seismic demands of the standard design for critical structures, systems, and components in Appendix 3B are bounded by the corresponding design certified seismic demands and, if not, the standard design for critical structures, systems, and components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands. Seismic demands investigated shall include forces, moments, deformations, in-structure response spectra, and seismic stability of the structures.	3.7.2

3.7.2.6 Conclusion

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

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Introduction

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

3.7.3.2 Summary of Application

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

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

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

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

3.7.3.3 Regulatory Basis

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

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

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

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

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

The guidance in DSRS Section 3.7.3, Revision 0, “Seismic Subsystem Analysis,” (ML15355A384) lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections. In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

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

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

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

3.7.3.4 Technical Evaluation

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

3.7.3.4.1 Seismic Analysis Methods

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

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

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

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

3.7.3.4.2 Determination of the Number of Earthquake Cycles

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

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

3.7.3.4.3 Procedure Used for Analytical Modeling

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

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

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

<u>NPM</u>

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

Fuel Storage Racks

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

Reactor Build Crane

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

Bioshields

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

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

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

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

3.7.3.4.4 Basis for Selection of Frequencies

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

3.7.3.4.5 Analysis Procedure for Damping

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

3.7.3.4.6 Three Components of Design Ground Motion

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

3.7.3.4.7 Combination of Modal Response

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

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

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

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

3.7.3.4.9 Multiple-Supported Equipment and Components with Distinct Inputs

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

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

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

3.7.3.4.10 Use of Equivalent Vertical Static Factors

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

3.7.3.4.11 Torsional Effect of Eccentric Masses

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

3.7.3.4.12 Seismic Category I Buried Piping, Conduits, and Tunnels

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

3.7.3.4.13 Methods for Seismic Analysis of Seismic Category I Concrete Dams

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

3.7.3.4.14 Methods for Seismic Analysis of Aboveground Tanks

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

3.7.3.5 Combined License Information Items

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

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

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

3.7.3.6 Conclusion

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

3.7.4 Seismic Instrumentation

3.7.4.1 Introduction

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

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

3.7.4.2 Summary of Application

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

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

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

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

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

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

- One SMA located on the RXB roof.

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

- One SMA located in the CRB in the MCR.

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

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

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

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

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

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

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

3.7.4.3 Regulatory Basis

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

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

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

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

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

3.7.4.4 Technical Evaluation

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

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

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

3.7.4.5 Combined License Information Items

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

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

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

3.7.4.6 Conclusion

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

3.8 Design of Category I Structures

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

3.8.1 Concrete Containment

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

3.8.2 Steel Containment

3.8.2.1 Introduction

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

3.8.2.2 Summary of Application

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

The summary of FSAR Section 3.8.2 is provided below:

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

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

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

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

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

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

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

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

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

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

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

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

3.8.2.3 Regulatory Basis

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

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

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

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

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

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

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

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

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

DSRS Section 3.8.2 lists the acceptance criteria to meet the above requirements, as well as review interfaces with other SRP sections. In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.7, Revision 3, "Control of Combustible Gas Concentrations in Containment," issued March 2007.

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

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

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

3.8.2.4 Technical Evaluation

The staff reviewed Section 3.8.2 and Chapter 6 of the FSAR against the 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 above.

DSRS Section 3.8.2 provides guidelines for the staff to use in reviewing the technical areas related to the design of the steel portion of the containment that is not backed by concrete, based on 10 CFR 50.55a; GDC 1, 2, 4, 16, and 50; 10 CFR Part 50, Appendix B; 10 CFR 50.44; and 10 CFR 52.47(b)(1). The staff used the guidance in DSRS Section 3.8.2 to review Section 3.8.2 of the FSAR. A summary of the application is discussed in SER Section 3.8.2.2 above.

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

3.8.2.4.1 Description of Steel Containment

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

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

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

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

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

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

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

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

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

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

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

In FSAR Table 6.2-4, the applicant lists the containment penetrations and states which penetrations are used for each process system fluids or gases. There are 45 penetrations in the CNV. Figures 6.2-2a, “Containment Vessel Assembly,” and 6.2-2b, “Containment Vessel Assembly,” in the FSAR show the CNV top head and side penetrations and listing the penetrations with nozzle numbers. The applicant describes types of penetrations as follows:

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

During the audit, the staff requested engineering documents for review to enhance understanding of the engineering methodologies and analysis details used in qualifying the CNV structure against the applicable codes and standards. The staff finds that the applicant provided sufficient information describing the CNV in the FSAR, and that the FSAR description complies with the acceptance criteria identified in DSRS 3.8.2.II.1.

3.8.2.4.2 Applicable Design Codes, Standards, and Specifications

The staff reviewed the codes, standards, and specifications in Sections 3.8.2.2 and 3.8.2.4 of the FSAR, against the list in DSRS Section 3.8.2.II.2. The applicant describes that the CNV

ASME, Class MC component is designed, constructed, and stamped as an ASME Code Class 1 vessel in accordance with ASME Code, Section III, Subsection NB, except that overpressure protection is in accordance with Article NE-7000 instead of Article NB-7000 of the ASME Code, Section III. The staff also audited the design specifications of the CNV, CNV support, and TSS provided in document EQ-105619, Revision 0, “ASME Design Specification for Containment Vessel and top Support Structures.”

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

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

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

3.8.2.4.3 Loading Criteria, Including Loads and Load Combinations

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

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

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

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

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

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

3.8.2.4.4 Design and Analysis Procedures

In FSAR Sections 3.8.2.4 and 6.2.1.1.2, the applicant provides an overview of the design and analysis requirements for the CNV and CNV supports. In FSAR Sections 3.8.2.1.2 and 3.8.2.2.2, the applicant describes that the CNV is an ASME, Class MC (steel) containment whose design, analysis, fabrication, inspection, testing, and stamping conform to the ASME Code, Class 1 pressure vessel requirements in accordance with Section III, Subsection NB as permitted by NCA-2134(c). In FSAR Table 3.8.2-1 “Design and Operating Parameters,” and Table 6.2-1, “Containment Design and Operating Parameters,” of the FSAR, the applicant provides the design and operation parameters of internal/external pressure and temperatures. Further, in FSAR Section 3.8.2.4, the applicant described that the CNV design and analyses (stress and fatigue usage) conform to the requirements of ASME Code, Section III, Subarticle NB-3200, the CNV support design and analysis conform to the requirements of Subarticle NF-3200, the fabrication conforms to the requirements of Article NB-4000 and Article NF-4000, and the nondestructive examination of pressure retaining and integrally attached materials meet the requirements of Article NB-5000 and Article NF-5000. The applicant also describes that the overpressure protection is performed in accordance with Article NE-7000 in the ASME, Section III.

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

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

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

3.8.2.4.5 Analysis Procedures

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

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

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

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

3.8.2.4.6 Containment Vessel Stress Analysis

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

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

During the audit, the staff requested numerous documents that provided the basis of construction of the CNV per the requirements of Section III of the ASME Code. The staff reviewed the documents and confirmed that the methodologies and the results presented are consistent with DSRS Acceptance Criterion of 3.8.2.II.4 and 5. The staff summary of its review of selected documents as related to the qualification of the CNV are as follows:

The applicant provides the design specifications of the CNV, CNV support and TSS in EQ-105619, Revision 0, "ASME Design Specification for Containment Vessel and top Support Structures." The applicant classifies the CNV as ASME Section III, Class MC. As permitted by NCA-2134(c), the applicant describes that the CNV is constructed and stamped as an ASME BPVC, Section III, Class 1 vessel in accordance with ASME BPVC, Section III, Subsection NB except that for the overpressure protection shall be in accordance with NE-7000.

The scope of the preliminary document, EC-124581, Revision A, "ASME Code Evaluation of the Lower CNV," is to provide analytical analysis of the lower sections of CNV, namely, lower shell, transition shell, core region shell, lower head of the CNC and the CNV support skirt. **[[**

]]. Based on the review, the staff finds that this preliminary analytical evaluation of the lower sections of CNV is acceptable since the results are within the allowable design limits in Subsections NB-3000, NF-3000 and Appendix XIII of Section III the ASME Code.

The scope of the document EC-140851, Revision B, “ASME Code Evaluation of CNV Head Feedwater Nozzle Region,” is to evaluate the analytical analysis of the CNV upper head containing the feedwater (FW) nozzle. **[[**

]]. Based on its review, the staff finds that this preliminary analytical evaluation of the CNV Head Feedwater Nozzle Region is acceptable since the results are within the allowable design limits in Subsections NB-3000 and Appendix XIII of the ASME Code, Section III.

The scope of the document EC-128779, Revision 0, “CNV Sizing Calculation,” is to determine the minimum pressure thickness of the CNV shell walls, nozzles, safe-ends and bolted cover plates using the design conditions (temperature, pressures, etc.), material properties, geometry of components. **[[**

]], the staff finds that the design calculations meet the requirements of Article NB-3324.1 of the ASME Code, Section III and are, therefore, acceptable.

The applicant evaluates the CNV top head shell on the effects of loading from the piping and electrical penetrations. The applicant checks buckling of the CNV for the Service Level D seismic event using the requirement of ASME Code Case N-759-2. The applicant also checks buckling on the inside knuckle regions of the top head and bottom head due to internal pressure causing compression in the knuckle regions that using hand calculation based on Equation 4.3-19 from Section VIII of ASME Code, Division 2. The staff finds that the top head shell and bucking checks are consistent with the acceptance criteria of DSRS 3.8.2.II.4.B, and therefore is acceptable.

The analytical evaluations for the CNV were partially available for staff review. Consequently, for its safety review, the staff also used information from the Chapter 3 SER (ML20205L491) for NuScale Design Certification Application (DCA), Revision 5 (ML20225A071). The staff noted that the changes to the geometry of the CNV and their integral components in SDAA are similar to those in the DCA. Therefore, the staff concluded that the external loading conditions on the SDAA CNV would be bounded by those in the DCA due to having smaller mass participation from the SDAA RXB structure and pool water.

The staff also noted that the primary loading conditions on the SDAA CNV are from the internal pressure, penetrations and support attachments. As discussed in Section 3.8.2.4.5 of this report, the applicant conservatively determines the ultimate internal pressure capacity of the CNV for a beyond design-basis LOCA is well over the internal design pressure.

Therefore, based on its review of the available and selected engineering documents, the staff finds that the applicant’s analysis results for the CNV are acceptable because they meet the Subsections NB and NF of Section III of ASME Code and are consistent with the acceptance criteria in DSRS 3.8.2.II.4 and 5.

3.8.2.4.7 Containment Vessel Lateral Support Lugs Analysis

In FSAR Section 3.8.2.4.2, “Containment Vessel Lateral Support Lugs,” the applicant describes the analytical stress evaluations of CNV lateral support lugs performed for the loads and load combinations described in FSAR Section 3.8.2.3 and Table 3.8.2-5. The applicant describes that the lateral support lugs are attached to the CNV upper shell as defined by Paragraph NB-1132.1(a) of Section III of the ASME Code. The lateral support lugs are laterally constrained by the NPM lugs attached to the NPM bay walls. The applicant provides the material specification of CNV support lug as UNS N06690 in Table 5.5.1, “Material for CNV,” of EQ-146988, “ASME Design Specification for Containment Vessel.” The applicant describes that the stress and fatigue results of the CNV lugs are evaluated in accordance with the limits in Subarticle NB-3200 of Section III of the ASME Code.

The applicant provides Figure 6.2-2a, “Containment Vessel Assembly,” in the FSAR showing that the CNV lateral lugs are located near the top of the CNV. The staff also reviewed the following drawings: SL-SPD1-00-M-GA-F010-05101, Revision 0, “Reactor Building General Section B-B,” and SL-SPD1-00-M-GA-F010-30001, Revision 0, “Reactor Building General Arrangement Plan View Elevation 85’-0”,” to gather configurational understanding of the CNV lugs. The staff finds the information provided in FSAR Section 3.8.2.4.2 is acceptable because the stress and fatigue results of the CNV lugs are evaluated in accordance with the limits in Subarticle NB-3200 of Section III of the ASME Code and is consistent with the acceptance criteria in DSRS Section 3.8.2.II.4

3.8.2.4.8 Containment Vessel Lower Support Analysis

In FSAR Section 3.8.2.4.3, “Containment Vessel Lower Support,” the applicant describes the analytical stress evaluations of the CNV skirt support performed for the loads and load combinations described in Section 3.8.2.3 and Table 3.8.2-5 of the FSAR. The applicant describes that the CNV support skirt is integrally welded to the bottom of the CNV lower head and provides vertical restraint, by bearing on the reactor pool floor, and horizontal restraint, by contact with a metal ring called the passive skirt support, which is attached to the reactor pool floor. The applicant classifies the CNV support skirt as an ASME Section III, Class MC Support. However, the applicant describes the design specifications of the CNV support skirt in document EQ-105619, Revision 0, and as permitted by NCA-2134(d), the CNV skirt support will be constructed as an ASME Section III, Class 1 support in accordance with ASME Section III, Subsection NF requirements. The applicant describes that the stress and fatigue evaluations of the CNV support skirt are performed in accordance with Subarticle NF-3200 and the support skirt is constructed in accordance with the requirements of Article NF-4000 of Section III of ASME Code.

FSAR Figure 3B-33 shows the general configuration of CNV lower support located on top of foundation. The staff also reviewed the following general arrangement drawings of SL-SPD1-00-M-GA-F010-05101, Revision 0, “Reactor Building General Section B-B,” and SL-SPD1-00-M-GA-F010-10001, Revision 0, “Reactor Building General Arrangement Plan View Elevation 25’-0”,” to gain configurational understanding of the CNV lower support. The staff finds the information provided in FSAR Section 3.8.2.4.3 is acceptable because the stress and fatigue results of the CNV lugs are evaluated in accordance with the limits in Subsection NF-3000 of Section III of the ASME Code and are consistent with the acceptance criteria in DSRS Section 3.8.2.II.4

3.8.2.4.9 Containment Vessel Reactor Pressure Vessel Supports Analysis

In FSAR Section 3.8.2.4.4, “Containment Vessel Reactor Pressure Vessel Supports,” the applicant describes the analytical stress evaluations of the CNV to RPV support for the loads and load combinations described in FSAR Section 3.8.2.3 and FSAR Table 3.8.2-5. In FSAR Table 6.1-2. the applicant provides the material specifications of CNV-RPV support ledge shell lug and RPV support ledge/gussets as SA-168, UNS N06690.

The applicant identifies the CNV-RPV support as Subsection NF, Class 1. The applicant describes that the RPV is laterally and vertically supported to the CNV at four locations by ledges that are integrally welded to the CNV inner and RPV outer shell surfaces. The applicant also describes that vertical lift off is prevented by a threaded pin and collar at the CNV-RPV ledge supports and the seismic connection at the lower RPV-CNV interface is only for lateral support in seismic events.

The applicant describes that the stress and fatigue results of the RPV-CNV support are evaluated in accordance with the limits in Subarticle NB-3200 of Section III of the ASME Code.

The staff finds that the information provided in FSAR Section 3.8.2.4.4 is acceptable because the stress and fatigue results of the CNV lugs are evaluated in accordance with the limits in Subarticle NF-3000 of Section III of the ASME Code and are consistent with the acceptance criteria in DSRS Section 3.8.2.II.4.

3.8.2.4.10 Containment Vessel Ultimate Capacity Analysis

In FSAR Section 3.8.2.4.5, “Containment Vessel Ultimate Capacity,” the applicant describes determination of the ultimate pressure capacity of the CNV to the guidance in Appendix A of NUREG/CR-6906 and the failure criteria of the CNV are based on guidance in RG 1.216. The applicant refers to the Technical Report, TR-121516, “CNV Ultimate Pressure Integrity,” (ML19158A382) describing the methodology, ultimate pressure, and method of failures for the CNV internal pressure capacity for a beyond design-basis LOCA.

The applicant determination of the ultimate pressure capacity is by meeting one of the following failure criteria:

A. A maximum global membrane strain away from discontinuities of 1.5 percent is reached.

B. Loss of bolt preload occurs at any bolted CNV opening.

C. Buckling occurs in the knuckle of the upper or lower CNV head due to internal pressure.

D. A flange gap that exceeds the calculated allowable values is reached at the outer O-ring of any bolted CNV opening.

E. Solution divergence occurs.

The applicant uses multiple finite element models and analyses to evaluate the bolted connections, shell regions away from concentrations, and buckling of the knuckle regions in the heads. The applicant determines the ultimate internal pressure capacity of the CNV, considering a conservative temperature level of 600 degrees Fahrenheit for the material properties, where the failure is at the CNV manway port due to the CNV shell pressure dilation that promotes gaps of the manway port opening. This ultimate internal pressure is above the internal design pressure of 1,200 psia as provided in Tables 3.8.2-1 and 6.2-1 of the FSAR.

Based on its review, the staff finds the applicant's design evaluation results acceptable because they are consistent with acceptance criteria in DSRS 3.8.2.II.4.F, and therefore, satisfy the criteria of GDC 50 and 10 CFR 50.44.

3.8.2.4.11 Containment Vessel Radiation Exposure Effects

In FSAR Section 3.8.2.4.6, "Containment Vessel Radiation Exposure Effects," the applicant describes the CNV radiation exposure effects. The applicant refers to FSAR Section 6.2 on the discussions on the effects of irradiation embrittlement of lower CNV materials, which concludes that loss of fracture toughness is negligible at the beltline SA-965 Grade FXM-19 base metal and associated weld metal from neutron irradiation during the design lifetime. Section 8.2.6, "Irradiation Embrittlement of Lower Containment Vessel," in TR-123952-NP, the applicant states that the peak 57 effective full-power years fluence is 0.0075 dpa for the lower CNV beltline base metal and 0.0035 dpa for the lower CNV beltline welds and concludes that loss of fracture toughness in the lower CNV beltline SA-965 Grade FXM-19 base metal or associated weld metal from neutron irradiation during the design lifetime is negligible as well as these peak fluence values are tiny fractions of NUREG/CR-7027 threshold fluence for irradiation embrittlement.

The staff performed an independent review of NUREG/CR-7027 and determined that the existing fracture toughness data on austenitic stainless steels irradiated in light-water reactors indicate little or no loss of fracture toughness below an exposure of about 0.5 dpa. The existing data for welds also suggest that ~0.3 dpa may be considered a threshold neutron dose below which irradiation has little or no effect on fracture toughness of stainless steels welds. Therefore, the staff confirms that the applicant's conclusion that *"loss of fracture toughness in the lower CNV beltline SA-965 Grade FXM-19 base metal or associated weld metal from neutron irradiation during the design lifetime is negligible,"* due to its distance from the core region and is consistent with the acceptance criteria in 3.8.2.II.4.C, therefore the staff finds this acceptable.

3.8.2.4.12 Containment Vessel Cyclic Fatigue Analysis

In FSAR Section 3.8.2.4.7, "Containment Vessel Cyclic Fatigue," the applicant performs the fatigue analysis for the CNV including for the Class 1 reactor coolant pressure boundary

(RCPB) nozzles penetrations (reactor cooling systems, pressurizer spray and reactor pressure vessel high point degasification) in accordance with the ASME Code, Section III, Paragraph NB-3200. Applicable cyclic, dynamic, pressure, and thermal transient loads and load combinations, discussed in Section 3.8.2.3, are considered in the fatigue evaluation. For the CNV process fluid penetrations classified as ASME Code Class 1, the fatigue analysis considers effects of the pressurized-water reactor (PWR) environment in accordance with RG 1.207 and NUREG/CR-6909. Section 3.7.3 of this report discusses operating basis earthquake seismic loads and analysis.

The staff finds that the calculations are in various stages of completion (some calculations were final with assumptions that needed to be verified, and some that were not final calculations) but determined that the calculations were in accordance with Section III of the ASME Code, and that any deficiencies were either identified in the corrective action program or were planned to be addressed when the calculations are made final during the ITAAC closure process. The staff finds that this satisfies GDC 50.

3.8.2.4.13 Structural Acceptance Criteria

In FSAR Section 3.8.2.5 the applicant describes the CNV structural integrity acceptance criteria limits, which are developed in accordance with Subarticles NB-3200 and NF-3200 of ASME Code, Section III, for plate-type and shell-type supports for the CNV support. In FSAR Tables 3.8.2-2, Table 3.8.2-3, and Table 3.8.2-5, the applicant tabulates the plant events, service levels load combinations and the ASME Code allowable stress limits for CNV pressure retaining items, Class 1 Supports, and Class 2 Supports for the CNV, and the associated bolts, bolted connections and supports. The applicant describes that the CNV is fabricated, installed, and tested according to Subsections NB and NF of ASME Code, Section III. The applicant refers to TR-123952, Revision 0, and is also provided in Section C of Section 6.2, of FSAR, describing CLIP. The CLIP provides assurance that leakage integrity of containment is maintained, and that containment leakage does not exceed allowable leakage rate values per the requirements of GDC 52 and performance of the preoperational and periodic integrated leak rate testing per the requirements of 10 CFR Part 50, Appendix J.

The staff reviewed FSAR Table 3.8.2-2 through Table 3.8.2-5, and finds them to be acceptable because the structural acceptance criteria comply with those identified in DSRS 3.8.2.II.5 and the loads and stress are defined in accordance with Section III of the ASME Code.

3.8.2.4.14 Materials, Quality Control Programs, and Special Construction Techniques

In FSAR Section 3.8.2.6 and Section 6.1.1.1, the applicant describes the CNV materials, which conform to the requirements of Subarticle NB-2000 and NF2000 in the ASME Code. The CNV fabrication conforms to the requirements of Subarticles NB-4000 and NF-4000 in the ASME Code. The CNV uses no special construction techniques. The quality control program involving materials, welding procedures, and nondestructive examination of welds conforms to Subarticles NB-2000, NB-4000, and NB-5000 in the ASME Code. FSAR Tables 6.1-1 and 6.1-2 list the materials used for fabrication of the CNV and attachments.

The staff reviewed FSAR Section 3.8.2.6, Section 6.1.1.1 and Tables 3.8.2-2 through Table 3.8.2-5, and finds to be acceptable because the materials, quality control and special construction techniques acceptance criteria comply with those identified in DSRS Section 3.8.2.I.6 and are in accordance with Section III of the ASME Code.

3.8.2.4.15 Testing and Inservice Inspection Programs

FSAR Section 3.8.2.7 describes the testing and ISI requirements for the CNV.

The applicant describes that nondestructive examination of the CNV pressure retaining and integrally attached materials meet the requirements of Article NB-5000 in the ASME Code, Section III, and NF-5000 using nondestructive examination methods of Section V and inservice inspections of Section XI.

The applicant describes the hydrostatic test of the CNV, performed after fabrication, in accordance with ASME Section III, Paragraph NB-6000 by pressurizing it to a minimum of 25 percent over design pressure of 1,200 psia. The acceptance criterion for the test is that there are no indications of leakage.

The applicant describes that nondestructive examination of the CNV after fabrication includes the following preservice examinations that are performed after hydrostatic testing but before code stamping:

- General visual examinations for pressure retaining surfaces above the reactor pool level in accordance with Paragraph IWE-2200,
- VT-3 visual examinations for pressure retaining surfaces below the reactor pool level in accordance with Paragraph IWE-2200,
- VT-1 visual examinations for pressure retaining bolting in accordance with Paragraph IWE-2200,
- Volumetric examinations for select welds in support of the break exclusion zone requirement in accordance with augmented requirements,
- Volumetric examinations for the: (a) CNV upper head to CNV upper seismic support shell, (b) CNV lower shell to CNV lower transition shell, and (c) CNV lower core shell to CNV lower head circumferential vessel welds in accordance with augmented requirements.

The applicant also refers to FSAR Section 6.2 that describes inservice inspections of the CNV. As described above, the applicant also refers to TR-123952, Revision 1, that describes the CLIP that provide adequate assessments of overall CNV leakage rates by performing Type B and Type C tests of 10 CFR Part 50, Appendix J.

The applicant also describes that Subsection IWE of the ASME Code, Section XI, requires for Class MC structures, 80 percent of one side of the pressure retaining boundary of the vessel be accessible for either direct or remote visual examinations for the life if the plant.

The staff reviewed FSAR Section 3.8.2.7 and finds that it is in accordance with the guidance in DSRS Section 3.8.2.II.7. The staff finds that this satisfies the criteria in GDC 53. SER Section 6.2.6 discusses containment leakage testing and gives the staff evaluation of compliance with 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.”

3.8.2.5 *Combine License Information Items*

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

3.8.2.6 *Conclusion*

The staff finds that the applicant has adequately addressed the design of the steel containment in the FSAR, as supplemented by the documents presented during the audit, in accordance with the acceptance criteria set forth in DSRS Section 3.8.2. On this basis, the staff concludes that the design of the steel containment is acceptable and meets the relevant requirements of 10 CFR 50.44, 10 CFR 50.55a, and GDC 1, 2, 4, 16, 50, and 53.

The staff notes that some of the documents presented during the audit are in preliminary stages (e.g., EC-140853, Revision B, "ASME Code Evaluation of CNV Head Feedwater Nozzle Region," EC-124581, Revision A, "ASME Code Evaluation of the Lower CNV," etc.), some of the completed documents are currently being revised (e.g.; EQ-105619, Revision 0, per the response in A-3.8.2-2), and some documents are yet to be performed (e.g., for the upper section of the CNV) requiring ITAAC closure. The staff performed reviews of these documents to better understand the methodologies and results utilized in the design of the steel containment, as well as to inform the staff's safety findings. The staff believes that the NuScale QA processes for these documents will be completed as required, and the documents will be made available for staff review as needed in the future. Based on the staff's review of the FSAR, as supplemented by the supporting documents as discussed above, the staff has reasonable assurance that the final versions of the supporting documents will meet the regulatory requirements committed in DSRS Section 3.8.2, and that the final design of the steel containment will be acceptable.

3.8.3 Concrete and Steel Internal Structures of Steel Containments

The NPM does not use internal structures (compartments, pedestals, or walls). SER Section 3.8.2 gives the staff's evaluation of connections between the CNV and the reactor vessel.

3.8.4 Seismic Category I Structures

3.8.4.1 *Introduction*

This section describes the review of areas relating to the structural design of Seismic Category I structures other than the containment, namely, the RXB and CRB. DSRS Section 3.8.4, "Other Seismic Category I Structures," provides guidelines and acceptance criteria for reviewing issues related to the design of Seismic Category I structures other than the containment.

The Seismic Category I structures are portions of the RXB and portions of the CRB. These buildings are site-independent and designed for the CSDRS and the certified seismic design response criteria - high frequency (CSDRS-HF) described in Section 3.7.1. The static and seismic analyses of the Seismic Category I structures are performed using ANSYS (Reference 3.8.4-1). The ANSYS software used for performing seismic analysis of SC-I SSCs conforms with the requirements for computer software as per the NuScale Quality Assurance Program Description (QAPD) (Reference 3.7.1-12).

3.8.4.2 Summary of Application

The applicant describes that the static and seismic analyses are performed using the ANSYS finite element computer code, which conforms to the applicant's requirements for computer software under the NuScale QAPD. The applicant also describes that the Seismic Category I buildings are site-independent and designed for the CSDRS and the CSDRS-HF as described in Section 3.7.1.

In FSAR Section 3.8.4.1, "Description of the Structures," the applicant describes the physical description and the primary functions of the Seismic Category I structures, primarily RXB, CRB and other structures that also includes RXB components. The applicant also issues a COL Item 3.8-1 for the design of the reactor flange tool.

In FSAR Section 3.8.4.2, "Applicable Codes, Standards, and Specifications," the applicant describes the codes, standards, and specifications meeting acceptance criteria in DSRS. Specifically, compliance to ACI 349, AISC N690, ASCE 7 and ASCE 4.

In FSAR Section 3.8.4.3, "Loads and Load Combinations," the applicant refers to Table 6-2 and Table 8-2 in TR-0920-71621-P-A, Revision 1, "Building Design and Analysis Methodology for Safety-Related Structures," (ML20353A404) (TR) of SC walls and RC structures, respectively, for the loading combinations for the structural design and analysis of the Seismic Category I portions of RXB and CRB.

In FSAR Section 3.8.4.4, "Structural Modeling and Analysis Procedures," the applicant describes that the methodology of structural design and analysis of the Seismic Category I portions of RXB and CRB provided in TR-0920-71621-P-A, Revision 1 and the calculated DCR values at selected critical sections are summarized in FSAR Appendix 3B.

In FSAR Section 3.8.4.5, "Structural Design and Acceptance Criteria," the applicant describes the acceptance criteria in TR-0920-71621-P-A, Revision 1.

In FSAR Section 3.8.4.6, "Materials, Quality Control and Special Construction Techniques," the applicant describes the material properties in Table 3.8.4-3 for the structural design and analysis.

In FSAR Section 3.8.4.7, "Testing and Inservice Inspection Requirements," the applicant only requires quality control performances for concrete members and SC wall per the requirements of ACI 349 and AISC N690 and issued a COL Item 3.8-2 for a site-specific program for monitoring and maintenance of the Seismic Category I structures.

In FSAR Section 3.8.4.8, "Evaluation of Design for Site-Specific Acceptability," the applicant describes the evaluation of design and analysis in Section 3.8.4 could be acceptable if site-specific parameters were to be shown less than provided in Table 2.0-1 of the FSAR and forces experienced at the critical sections in Seismic Category I structures under the site-specific earthquake are less than that provided in the FSAR and supporting reports.

In FSAR Appendix 3B, "Design Reports and Critical Section Details," the applicant summarizes the structural design and analysis with of the RXB and CRB that includes selection criteria for the critical sections, checking for the structural integrity of critical sections DCR values under load combinations.

ITAAC: ITAAC are evaluated in SER Section 14.3.

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

Technical Reports: There are no Technical Reports for FSAR Section 3.8.4.

3.8.4.3 Regulatory Basis

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

- 10 CFR 50.55a(1)(i)(E)(17) and GDC 1, as they relate to SSCs being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 2, as it relates to the design of structures important to safety being capable to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, without loss of capability to perform their safety functions, the design bases for these structures should reflect as appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

- GDC 4, as it relates to appropriately protecting structures important to safety against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

- GDC 5, “Sharing of Structures, Systems, and Components,” as it relates to not sharing structures important to safety among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

- 10 CFR Part 50, Appendix B, as it relates to the QA criteria for safety-related SSCs of nuclear power plants.

The guidance in DSRS Section 3.8.4 lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other DSRS and SRP sections. In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.69, Revision 1, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants,” issued May 2009.

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

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

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

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

- RG 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," issued May 2012.

- RG 1.199, "Anchoring Components and Structural Supports in Concrete," issued November 2003.

- RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," issued October 2011.

3.8.4.4 Technical Evaluation

The staff reviewed FSAR Section 3.8.4 in accordance with the DSRS Section 3.8.4. DSRS Section 3.8.4 describes the acceptance criteria to meet the relevant requirements of the NRC's regulations pertaining to the structural design of Seismic Category I structures other than the containment. The summary of the application is discussed in SER Section 3.8.2.2 above.

3.8.4.4.1 Description of the Structures

In FSAR Sections 1.2.2.1, 3.7.2.1.2.1, 3.8.4.1.1 and Appendix 3B, the applicant provides general information related to the RXB. The RXB is deeply embedded with a center of gravity below the site grade elevation. The overall dimensions of the RXB are 231.5 ft, 155.5 ft, 171 ft in the east–west (X), north–south (Y), and vertical (Z) directions, respectively, and consists of SC walls, RC basemat and slabs and are designed to withstand the effects of natural phenomena. The thickness of the main structural interior and exterior SC walls is 4 ft. The RC floor slabs are either 2 ft. or 3-ft thick. The thickness of the east and west exterior SC walls is 5 ft. The thickness of the north and south exterior walls is 4 ft. The thickness of the basemat foundation is 8 ft. The thickness of roof slab is 3 ft.

The primary feature of the RXB is the pool located at the center of the building designed to be the UHS for the NPMs. The pool consists of the spent fuel pool and the refueling pool housing up to six NPMs. The normal depth of the reactor pool water is maintained at 53 ft. RXB includes the following components: bioshields, RXB pool liner, equipment door and dry dock gate. The design properties of the critical SC wall and RC slab sections of the RXB are provided in Tables 3B-1 and 3B-2 of the FSAR.

In Section 3.8.4.1.10, of the FSAR, the applicant describes that the modular construction techniques will be used to construct the Seismic Category I RXB SC walls. Modular construction techniques increase the efficiency of construction and productivity because the steel portions of the SC walls are fabricated off-site in a controlled environment. Furthermore, construction of the formwork at the site is not required and reinforcement is not needed.

In FSAR Sections 1.2.2.2, 3.7.1.2.5, 3.8.4.1.2 and Appendix 3B, the applicant describes the CRB. The CRB is an RC structure comprised of Seismic Category I, Category II and Category III (per FSAR Table 3.2-1) sections. The overall dimensions of the CRB are 120 ft, 55 ft, and 50 ft in the east–west (X), north–south (Y), and vertical (Z) directions, respectively, and consists of RC basemat, walls and slabs and are designed to withstand the effects of natural phenomena. The primary function of the CRB is to house the MCR and the technical support center. The critical wall and slab sections of the CRB are provided in Tables 3B-3 and 3B-4, of the FSAR.

The staff reviewed the descriptions of structures in FSAR Sections 1.2.2, 3.7.2, 3.8.4 and Appendix 3B, including general arrangement drawings with plan and section views of the structures, overall structural dimensions, floor and wall thicknesses, floor elevations, and steel reinforcement configurations. The staff's review found the level of detail with respect to the description of structures is sufficient for defining the primary structural aspects and elements that are relied upon to perform the safety-related functions of these structures. Specifically, based on the structural descriptions provided in the FSAR, the staff was able to identify the structural load path for the transfer of loads from the roof to the basemat of the structures. Further, the staff was able to identify enough dimensions to develop the dynamic models for the seismic analyses of the structures and establish the relationship between adjacent structures. Additionally, the staff found the structural descriptions contained sufficient details to confirm the consistency of the structural design aspects (e.g., structural member capacities and reinforcement configuration) in the design descriptions with the reference design codes. Based on the above, the staff concludes that the descriptions of structures in the FSAR are acceptable.

3.8.4.4.2 Applicable Codes, Standards, and Specifications

FSAR Section 3.8.4.2 lists the codes, standards, and specifications applicable to the Seismic Category I portions of the RXB and CRB. The staff reviewed the list of codes, standards, and specifications to confirm that the criteria used in the analysis, design, and construction of the RXB and CRB are consistent with the established criteria, codes, standards, and specifications acceptable to the staff. DSRS Section 3.8.4.II.2 lists the codes, standards, and specifications acceptable to the staff.

Based on the applicant's use of codes, standards, and specifications consistent with DSRS Section 3.8.4.II.2, and the conservative implementation of AISC N690-12 as described above, the staff concludes that the information in FSAR Section 3.8.4.2 on applicable codes, standards, and specifications for the other Seismic Category I structures of the NuScale design is acceptable.

3.8.4.4.3 Loads and Load Combinations

In FSAR Section 3.8.4.3, the applicant describes the loads and load combinations for the RXB and CRB structural design conform to the ACI 349-13, endorsed by RG 1.142, and AISC N690-18, Appendix N9, endorsed by RG 1.243, as the basis for the loads and load combinations. In Sections 3.8.4.3.1 through 3.8.4.3.22 of the FSAR, the applicant provides symbols and detail description of the applicable loads. The applicant refers to Table 6-2 and Table 8-2 in TR-0920-71621-P-A, Revision 1, for the load combinations of the SC walls and RC structures.

The staff reviewed that Table 6-2 and Table 8-2 load combinations in the TR-0920-71621-P-A, Revision 1, and determined them to be consistent with Chapter NB2.5 of Specification N680-18 for the load resistance factor design (LRFD) method and ACI 349-13. For the differences in load factors between RG 1.142 and Table 8-2 in TR-0920-71621-P-A, Revision 1, the applicant selected the load factors in ACI 349-13 instead of the revised values in RG 1.142 because the values in ACI 349-13 more closely align with the recent design codes for developing load combinations.

The load combinations listed in ACI 349-13, and corresponding labels used in the calculations, are used in the design of the RC members and SC walls of the RXB and are provided in FSAR Tables 3B-6 and 3B-7, respectively. Further, DSRS provides the guidance and acceptance criteria specifically related to the design and evaluation of structural steel and concrete structures in the NuScale nuclear power plants. DSRS 3.8.4 refers to the applicable codes and standards, loads and load combinations, design and analysis procedures, structural acceptance criteria, and materials and special construction techniques. The acceptance criteria in DSRS 3.8.4-II states the structural acceptance criteria refers to the applicable codes of AISC N690-1994 and ACI 349-13 for the SC walls and concrete structures with additional criteria provided in RG 1.142. DSRS 3.8.4, Section II.4.J, also provides reference to guidance contained in NUREG/CR-6486, "Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants," and other applicable industry documents related to the use of modular construction methods. Use of these guides and specifications in TR-0920-71621-P-A, Revision 1, provide assurance and impose specific restrictions to ensure that the SC walls and concrete structures will perform their intended safety function with the identified loads and their load combinations, and therefore acceptable to the staff.

In FSAR Appendix 3B, Section 3B.1.3, the applicant determines the cracking states due to in-plane-demands using seismic combination of "D + 0.8L + E_{ss}," where a label of "CrkEs," was assigned for this load combination case in the calculations. Further, the staff agrees with this seismic load combination, since the maximum differential pressure load, Pa, fluid pressure load, F, soil pressure load, H, and thermal loads during normal and abnormal conditions, T_o, and T_a, are expected to have a major effect on the out of plane flexure of walls and slabs but only a minor effect in their in-plane direction.

The load combination in Equation 3B-1 is only used to evaluate the state of cracking in walls and slabs of the seismic force resisting system. The full load combinations in ACI 349 or AISC N690-18 are used to obtain the member forces for design.

In accordance with COL Item 3.6-1, the COL applicant will address final piping layout, analysis, and additional protection features as necessary. Based on the applicant's generic evaluation, the staff's review, and the site-specific verifications to be performed by the COL applicant, the staff finds the applicant's consideration of Yj, Yr, and Ym concentrated local load effects in the RXB design to be acceptable.

The load and load combinations follow the requirements of ASIC N690-18, Appendix N9, and 360-16, and consistent with the structural acceptance criteria in NRC NuScale DSRS Section 3.8.4, DSRS 3.7.2. DSRS 3.8.4 provides guidance regarding basic specifications for concrete and steel structures in compliance with NRC regulations and cites certain RGs and industry consensus codes and standards, specifically ACI 349-13 and AISC N690-18, Appendix N9 that are acceptable to the staff. The staff reviewed and compared the loads and load combinations presented in the FSAR with the referenced codes. The staff's review found that the load definitions and load combinations conform with the referenced codes and therefore acceptable.

3.8.4.4.4 Design and Analysis Procedures

In FSAR Section 3.8.4.4, and Appendix 3B, the applicant provides an overview of the design and analysis requirements for the RXB and CRB. The applicant describes that the structural integrities of RXB and CRB are to be maintained, and the safety-related SSC remain operable during and following an earthquake represented by the CSDRS or the certified CSDRS-HF. Specifically, Appendix 3B addresses twelve critical sections in the RXB and 5 in the CRB that were selected because they (1) perform a safety-critical function, (2) are subjected to large stress demands, (3) are considered difficult to design or construct, or (4) are representative of the structural design.

3.8.4.4.5 Analysis Procedures

The applicant performed static and seismic analyses with the ACS SASSI, SHAKE2000 and ANSYS computer codes to determine the structural response to non-seismic and seismic loads, including the fluid-structure interaction effects. Additionally, the applicant performed thermal and pressurization analyses with ANSYS. Consistent with the acceptance criteria in DSRS Section 3.8.4.II.4, the staff determined the use of these computer programs to be acceptable because these programs are recognized and have a sufficient history of use in the nuclear industry to justify their applicability. Section 3.7.2 of this SER provides specific details with respect to the staff's review of the modeling and analysis performed with ACS SASSI and SHAKE2000.

ANSYS is a general purpose commercially available finite element program that has been widely used and accepted by the engineering community. It is used in a variety of structural applications including both linear and nonlinear static and dynamic analysis. Further, the engineering community has accepted ANSYS and used it in nuclear applications to obtain results that are also acceptable to the staff. The applicant performed the analyses with ANSYS, which consistent with the acceptance criteria in DSRS Section 3.8.4.II.4, the staff determined the use of these computer programs to be acceptable because these programs are recognized in the public domain and have a sufficient history of use to justify their applicability.

The design and analysis methodologies to determine the in-structure response spectra, effective stiffness, damping ratio, SC walls and connections and RC structures for Seismic Category I and Category II structures are described in the TR-0920-71621-P-A-R1 (TR). The staff reviewed the TR to confirm that the methodologies presented in the TR are consistent with the established criteria acceptable to the staff, as presented in Sections 3.7.2 and 3.8.4 of the DSRS.

The seismic load-resisting RC members and SC wall sections are checked for potential cracking from the CSDRS motion using the maximum force calculated in a structural member during the entire time-history. The method used to determine the effective stiffness and damping of the RC and SC members are recommended by ASCE/SEI 4-16, ASCE/SEI 43-19, and AISC N690-18 and are considered acceptable by the staff.

All cracked RC members are assigned the effective stiffness values and corresponding damping, expressed as fraction of critical damping, in accordance with ASCE/SEI 4-16

(Tables 3-1 and 3-2) and Table 3-1 of ASCE/SEI 43-19 and consistent with DSRS Section 3.7.2. All SC members are assigned the effective stiffness values as provided in Specification N690-18 and the damping values provided in ASCE/SEI 43-19.

As presented in the TR, the effective stiffness of an SC wall for both operational and thermal conditions is specified in Section N9.2.2 of AISC N690-18, and the out of plane flexural stiffness is calculated based on the stiffnesses of the cracked concrete infill and the faceplates using Equation A-N9-8 from Section N9.2.2 of Specification N690-18. The effective in-plane shear stiffness per unit width of the SC wall for operating conditions depends on the ratio of the average in-plane required shear strength, S_{rxy} and the concrete cracking threshold, S_{cr}, and is calculated by the trilinear relationship given by Equations A-N9-9 through A-N9-14 of Section N9.2.2 of AISC N690-18. The effective in-plane shear stiffness per unit width of the SC wall for accidental thermal conditions is determined using Equation A-N9-12 assuming cracked concrete. The threshold for crack developing in concrete, S_{cr}, is given by Equation A-N9-10.

This proposed methodology uses three different materials to represent the widely different surrounding media: a soft soil profile (Type 11), a rock profile (Type 7), and a hard rock profile (Type 9).

The analysis starts with the structure subjected to a CSDRS motion (demand) with the structural members having uncracked material properties and with response level- (RL) 1 damping values, as permitted by ASCE/SEI 43-19. This harmonic analysis is repeated for the three soil types considered. The seismic load-resisting RC members and SC wall sections are checked for the potential for cracking from the CSDRS motion using the maximum force calculated in a structural member during the entire time-history considering the most critical seismic load combination. The state of cracking of a member is calculated by the DCR.

The ISRS at a given location of a structural member is generated from the harmonic analysis with the updated stiffness and damping properties for each of the CSDRS motions by algebraic summation of the acceleration time history in each direction from the input motion in the three orthogonal X, Y, and Z directions. The ISRS is calculated at 2, 3, 4, 5, 7, and 10 percent of the critical damping. The average ISRS is calculated from the results obtained for each CSDRS motion used. The peak of the ISRS is broadened by ±15 percent following RG 1.122 (ML003739367) to account for the uncertainties in the structural frequencies. The ISRS, calculated at all nodes of a structural member, is enveloped by repeating the analysis for the three soil types selected, with the final ISRS selected that envelope the ISRS determined for each of the soil type.

In FSAR Section 3.8.4.3.22.2, the applicant describes the structural analysis performed for six NPMs are in their respective bays even though, the operations with fewer than six NPMs is allowed. However, the applicant describes that the dynamic effects on the building with fewer than six NPMs would be similar compared to when all six NPMs are in place. Consideration of all six NPMs is conservative given that the weight of all six NPMs in the pool would result in a relatively higher demand on the structural members of the RXB and, therefore, is acceptable.

In FSAR Section 3.8.4.4, and Appendix 3B, the applicant describes the structural modeling and analysis procedures for the RXB and CRB as well as associated components. The applicant uses the ANSYS structural analysis software for design and analysis of Seismic Category I and II structures. The applicant developed three-dimensional DB models of RXB and CRB, and CRB with major components using the following elements in ANSYS: solid shell (SOLSH190), shell

(SHELL181), beam (BEAM188), fluid (FLUID30), surface (SURF154), mass (MASS21), soil (MATRIX50), and surface elements (TARGE170 and CONT174). The applicant also used the thermal shell elements. SHELL131, for thermal analysis of the RXB SC and RC walls and slabs. Figure 3.7.2-60 through Figure 3.7.2-87 in the FSAR, show isometric and section views of the RXB ANSYS model identifying the primary element types with colors. Figure 3.7.2-4 in the FSAR, show isometric view of the CRB ANSYS model. The applicant used the hybrid DB RXB and RWB) and CRB models for design of the critical sections. The applicant used results from the ANSYS analysis to determine the structural response for static and dynamic loads, and post-processing of the analysis results. In addition to Soil 7, 9, and 11, soil 7 with soil separation modeling capability are considered in design calculations. In this context, soil 7 with soil separation capability is treated as a new soil type.

The staff finds the design-basis demands, which envelops analysis results considering a range of key structural and site parameters, and further site-specific verifications to be performed by the COL applicant to be conservative and acceptable.

3.8.4.4.6 Design Procedures

In FSAR Appendix 3B, the applicant describes the structural design and analysis of the RXB and CRB. Appendix 3B provides detail descriptions of the analysis and design of selected critical sections of the SC walls and RC sections to withstand the design-basis demands. The applicant describes the critical sections as part of a structure which are selected using the following criteria: (1) perform a safety-critical function, (2) are subjected to large stress demand, (3) are considered difficult to design or construct, or (4) are representative of the structural design. The applicant lists the critical sections of the RXB and CRB and their design properties in Tables 3B-1 and 3B-2, and in Tables 3B-3 and 3B-4 of the FSAR, respectively. FSAR Table 3B-8 and Table 3B-9 provide the cracked states based on load combinations for SC walls and RC members, respectively for the RXB. For each load combination and action, the applicant calculates the DCR values by dividing the total demand by the capacity of the critical sections.

The staff's review confirmed that the design conditions and calculated maximum DCR values of the critical sections as listed in FSAR Appendix 3B, Tables 3B-10, 3B-12, 3B-14 and 3B-20 for the RXB and in Tables 3B-23 through 3B-33 for the CRB are primarily equal to or less than 1.0. The DCR values are calculated and assessed following both element-based and panel section-based approaches, and SC wall regions where the element-based approach results in high DCR values are reevaluated using panel section-based approach. The applicant summarizes the maximum DCR values at critical section of SC Walls on RXB in Table 3B-10 showing the DCR counter plots in Figures 3B-17 through Figure 3B-22 and determines that there are three (3) localized areas at SC Wall RX1, RXE and Pool Wall where the DCR values exceed 1.0 by no more than 5% for design conditions of cMS (1.05 and 1.04) and Vy (1.02), respectively. The applicant describes that these localized exceedances occur at joint regions and reentrant corners and do not impact the overall structural design and safety of the walls. Because the effect of additional localized reinforcement is not included in the DCR calculations with the limitations of finite element modeling of challenging geometries, with elastic properties, providing high levels of localized stress levels. On this basis, the staff concludes that the DCR values of the selected critical sections of the RXB and CRB are within the limits specified by AISC N690-18 and are, therefore, acceptable.

3.8.4.4.7 Design Checks

Although, the staff performed independent reviews of all the design checks for the selected critical sections of RXB and CRB addressed in the FSAR, a summary of the evaluations of the design checks limited only to the sections in the RXB is described in this SER:

3.8.4.4.8 RXB SC Wall Design Checks

In Section 3B.2.2.3, of the FSAR, the applicant describes the procedures used to calculate the DCR values for each design condition at each of the finite elements of the critical SC wall sections for all active load combinations. The applicant lists the maximum DCR values in FSAR Appendix 3B, Table 3B-10 for the SC wall of RX1, RX4, RX4.3, RX 4.6, RXB, RXE and Pool Walls.

Based on its review, the staff finds the applicant's design evaluation results are acceptable because the RXB SC walls retain greater capacity than the demands. As concluded above, the staff also agrees with the applicant's conclusion that the three (3) localized DCR exceedances of no more than 5% would not affect the overall structural design and safety of RXB SC Walls. Based on the seismic demands and demonstration of adequate RXB SC Wall capacities, the staff concludes that the RXB SC walls are designed to retain their structural integrity when subjected to the design-basis demands and is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.9 Reinforced Concrete Slabs Design Checks in RXB

In FSAR Appendix 3B, Section 3B.2.3.1, the applicant identifies the critical sections from the member of RC slabs of the basemat, floor slab at elevation 100 ft, and the roof slab of the RXB. The applicant also identifies the initial design properties of the critical sections of RXB in Table 3B-2.

In FSAR Appendix 3B, Section 3B2.3.2, the applicant calculates the strength required at the critical sections of the RC slabs for the load combination of Load Combination 6 (LC6), as described in the FSAR Equation 3b-15, for Soil Type 7 (a rock profile). The peak contour plots of "Combined Demands for Load Combination LC6_p (force unit kip/ft and moment unit kip-in./ft)," of all elements for the basemat slab, floor slab at elevation 100 ft, and the roof slab under the LC6 demand load combination are shown in Figures 3B-24 through 3B-26. The element-based contour plots confirm that the selected critical sections match well with the locations where the demand values are the largest. Based on the contour plots, additional critical sections are selected for analysis under LC6.

FSAR Appendix 3B, Section 3B2.3.3, the applicant describes the design calculations of the RC members under demand values at the selected critical sections are shown and labeled in Figures 3B-27 through 3B-29 for the basemat slab, floor slab at elevation 100 ft and the roof slab. Using the initial design properties from Table 3B-1, the applicant calculates design values for out of plane demands (axial force - out of plane moment and axial force - out of plane shear).

The applicant also describes that the DCR values for in-plane shear conditions at the critical sections associated with the maximum allowed in-plane shear capacity are calculated for the governing load combination and soil type by taking the ratio of the required in-plane shear reinforcement, r_o, to the maximum in-plane shear reinforcement, r_{o_max}. The applicant describes that the additional required in-plane shear reinforcement are added to the longitudinal reinforcement. The final design properties of critical sections of RC members in RXB and the maximum DCR values are provided in Table 3B-14.

The applicant provides the summary of calculated DCR values at the critical sections of RC members for out of plane design conditions in FSAR Appendix 3B, Table 3B-13 and in-plane-shear design condition in Table 3B-13 that the DCR values are less than 1.0. The applicant provides the final design properties of critical sections of RC members in RXB and the maximum DCR values in Table 3B-14 that refers to Figures 3B-30 through 3B-31a providing reinforcement layouts for the basemat slab, floor slab at elevation 100 ft and the roof slab.

Based on its review, the staff finds the applicant's design evaluation results acceptable because, the DCR results for the reinforced concrete slabs in RXB listed in FSAR Appendix 3B, Tables 3B-12, 3B-13, and 3B-14, retain greater capacity than the design demands. Based on the seismic demands and demonstration of adequate structural capacity, the staff concludes that the reinforced concrete slabs in RXB are designed to retain their structural integrity when subjected to the design-basis demands and is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.10 NuScale Power Module Skirt Support

In FSAR Section 3B.2.4.1, the applicant describes NPM skirt restraint providing lateral support at the base of the NPM. Figure 3B-33 shows a plan view of the typical the NPM skirt support, the annular bearing plate, and lateral skirt restraints.

The applicant describes that the two evaluations: (1) the vertical analysis evaluates the annular bearing plate and (2) the lateral analysis evaluates the lateral braces based on the load path, starting with combined axial and bending of the braces, local bearing on the braces, evaluation of the brace connections to the SC walls, and local evaluation of the SC walls.

The applicant describes that the acceptance criteria of structural steel components of NPM skirt support skirt restraint confirm to the provisions of AISC N690-18. The LRFD load combinations using the governing equation of NB2-6 of AISC N690-18 is used for evaluation of the NPM skirt support skirt restraint.

The applicant tabulates the calculated DCR values of the components of NPM skirt restraints in Table 3B-15 for vertical and lateral analyses.

Based on its review, the staff finds the applicant's design evaluation results acceptable because, the DCR values for the NPM skirt restraint components listed in Table 3B-15, of the FSAR, retain greater capacity than the design demands. Based on the seismic demands and demonstration of adequate structural capacity as described above, the staff concludes that the lug restraints are designed to retain their structural integrity when subjected to the design-basis demands and is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.11 NuScale Power Module Lug Restraints

In FSAR Section 3B.2.4.2, the applicant describes that the three-lug restraint are Seismic Category I components where the configurations is extending from the bay SC walls with the wedge-jacks preventing the lateral movement of the NPM. FSAR Appendix 3B, Figures 3B-36 and 3B-37 show the plan layout of lug restraint configurations in six NPM bays and components of the typical lug restraint in one NPM bay, respectively. When the wedge-jacks in retracted position, there is a gap to allow the NPM to be removed/reinstalled by the RBC.

The applicant describes that the NPM lug restraint and associated components are evaluated for circumferential, radial, and vertical design seismic force from NPM lug. The design seismic loads are: Circumferential: 1,500 kips, Radial: +/- 112.5 kips and Vertical: +/- 112.5 kips. The applicant also notes that the boundary conditions NPM lug restraint is only subject to dead load and seismic loads at accident temperature. And the LRFD load combinations in Section NB2 of AISC N690-18 are used for evaluation of the NPM seismic lug restraint components. The applicant describes that the acceptance criteria of structural steel components of NPM lug restraint confirm to the provisions of AISC N690-18. In FSAR Appendix 3B, Table 3B-16, the applicant lists the calculated DCR value for the components in load path for the NPM lug restraint.

Based on its review, the staff finds the applicant’s design evaluation results acceptable because, the DCR values for the NPM Lug restrain components in the load path, listed in FSAR Table 3B-16, retain greater capacity than the design demands. Based on the seismic demands and demonstration of adequate structural capacity as described above, the staff concludes that the lug restraints are designed to retain their structural integrity when subjected to the design-basis demands and is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.12 Reactor Building Crane Corbel

In FSAR Appendix 3B, Section 3B.2.4.2, the applicant describes the RBC corbel, as shown in Figures 3B-23, 3B-38 and 3B-39, are the two continuous stiffened ledges attached to the RXB and RXD SC walls design at elevation 145 feet 6 inches to support moving point loads of RBC. The components of RBC corbel attached to the SC wall module are made from American Society for Testing and Materials (ASTM) A572 Grade 55 material. The loads evaluation of the RBC corbel is split into three sections: downward load analysis, upward load analysis, and lateral analysis. The LRFD load combinations of the governing equation of NB2-6 of AISC N690-18, corresponds to the extreme environmental load combination, is used for the structural integrity evaluations. In Table 3B-7 of the FSAR, the applicant lists the calculated DCR values for the components of RBC corbel.

Based on its review, the staff finds the applicant’s design evaluation results acceptable because, the DCR values of RBC corbel components in the load path listed in Table 3B-16, of the FSAR, retain greater capacity than the design demands. Based on the seismic demands and demonstration of adequate structural capacity as described above, the staff concludes that the RBC corbel components are designed to retain their structural integrity when subjected to the design-basis demands and is consistent with acceptance criteria in DSRS Section 3.8.4.II.2, and therefore, acceptable.

3.8.4.4.13 Structural Acceptance Criteria

The applicant describes that Seismic Category I structural steel and SC wall components are designed to AISC N690-18 and Seismic Category I SC members are designed to ACI 349-13. In FSAR Section 3.8.4.5, the applicant refers to TR-0920-71621-P-A, Revision 1, providing the structural design and acceptance criteria for the seismic Category RXB and CRB. The applicant also describes that the acceptance criteria in FSAR Appendix 3B provides with structural design evaluation results for selected critical sections of the RXB and CRB checked by calculating the DCR values, for each load combination and action, by dividing the total demand by the capacity, at the critical sections that the DCR value must be less than one for acceptable design.

The staff reviewed the structural acceptance criteria in FSAR Section 3.8.4.5 for application to the Seismic Category I structures, SC walls and RC members. The staff found the use of these structural acceptance criteria to be in accordance with the guidance given in DSRS Section 3.8.4.II.5 and, with respect to the updated criteria in AISC N690-18 and ACI 349-13 to be implemented conservatively as described in Section 3.8.4.2 of this report. On this basis, the staff finds the information in FSAR Sections 3.8.4.5 and Appendix 3B on the structural acceptance criteria to be acceptable.

3.8.4.4.14 Materials, Quality Control, and Special Construction Techniques

In FSAR Sections 3.8.4.6.1, the applicant refers to FSAR Table 3.8.4-3 for the principal construction materials for structures including SC walls are concrete, reinforcing steel, structural steel, stainless steel, bolts, anchor bolts, and weld electrodes. In FSAR Appendix 3B, Sections 3B.2.1.2, 3B.2.5.2 and 3B.3.1.2, the applicant provides the structural material requirements for RXB, SC Wall to RC slab connections and CRB, respectively. In FSAR Section 3.8.4.6.1.1, the applicant describes the structural concrete, used in the Seismic Category I RXB and CRB, conforms to ACI 349-13 as applicable, ACI 318-08 and for the SC walls per the requirements of AISC N690-18 and, as applicable, AISC 360-16.

Section 3.8.4.6.1.1, of the FSAR, the applicant provides the following engineering requirements for the concrete:

- The compressive strengths of concrete (f_c) are 5,000 psi, typical, and 7,000 psi, for the RXB roof and floor slabs, as tabulated in Table 3.8.4-3 of the FSAR.

- Concrete mixes are designed in accordance with ACI 211.1, “Selecting Proportions for Normal-Density and High Density-Concrete – Guide.”

- Cement conforms to the requirements of ASTM C150, “Standard Specification for Portland Cement.”

- Aggregates conform to the requirements of ASTM C33, “Standard Specification for Concrete Aggregates.” Further, ASTM C1260, “Standard Test Method for Potential Alkali Reactivity of Aggregates,” and C1293, “Standard Test Method for Determination of Length Change of Concrete Due to Alkali-Silica Reaction,” are used in testing aggregates for potential alkali-silica reactivity. Concrete with potentially reactive aggregates uses low-alkali cement.

- Air-entraining, chemical, and fly ash and pozzolan admixtures, if used, conform to the requirements of ASTM C260, “Standard Specification for Air-Entraining Admixtures for Concrete,” C494, “Standard Specification for Chemical Admixtures for Concrete,” and C618, “Standard Specification for Coal Fly Ash and Raw or Calcined Natural Pozzolan for Use in Concrete,” respectively.

- Water and ice for mixing are clean, with a total solids content of not more than 2,000 parts per million.

Further, in addition to ACI 349-13, Section 3.8.4.6.1.1, addresses codes and standards used for concrete construction, including placement, inspection, and testing. These include:

- ACI 301, “Specifications for Structural Concrete for Buildings,”

- ACI 304R, “Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete,”

- ACI 305.1, “Specification for Hot-Weather Concreting,”

- ACI 306.1, “Specification for Cold-Weather Concreting,”

- ACI 347, “Recommended Practice for Concrete Formwork,”

- ACI SP-2, “Manual of Concrete Inspection,”

- ASTM C94, “Specification for Ready-Mixed Concrete.”

Section 3.8.4.6.1.2, of the FSAR, the applicant describes the reinforcing steel conforms to ASTM-designation A615 grade 60 or A706 grade 60. Concrete reinforcement is emplaced in accordance with ACI 349. Reinforcing development length and splice length are calculated by ACI 349-specified formulas. Welded wire fabric for concrete reinforcement conforms to ASTM A185 (plain wire) or ASTM A497 (deformed wire).

Section 3.8.4.6.1.2, of the FSAR, the applicant refers to TR-0920-71621-P-A, Revision 1, for the SC wall requirements.

Section 3.8.4.6.1.4, for the FSAR, the applicant provides the following list of engineering requirements for the connections:

- Steel bolts conform to either ASTM A307, high-strength ASTM A490, or ASTM A325.

- Material. Steel studs meet the requirements of ASTM A108 and American Welding Society D1.1/D1.1M, “Structural Welding Code-Steel.”

- Anchor bolts are of type ASTM F1554 36 ksi or 55 ksi yield-strength material or ASTM F1554 105 ksi yield-strength or higher strength material.

- Welding electrodes are E70XX, unless otherwise noted on drawings, or are within the specification for ASTM A36 steel and E308L-16 or equivalent for ASTM A240-type 304-L stainless steel.

Section 3.8.4.6.1.4, for the FSAR, the applicant describes that grating is welded and galvanized steel, "Metal Bar Type," conforming to ANSI/NAAMM MBG 531-00, "Metal Bar Grading Manual," and ANSI/NAAMM MBG 532-00, "Heavy Duty Metal Bar Grating Manual." Further, the applicant describes that there are no safety-related reinforced masonry walls in Seismic Category I structures.

In Section 3.8.4.6.2, of the FSAR, the applicant refers to Chapter 17, for the details of the Quality Assurance Program.

The staff's review confirmed that the material specifications discussed above are within the scope of the primary design codes; that is, ACI 349-13 and AISC N690-18 or other referenced codes and standards are within the scope of the primary design codes as well as is consistent with DSRS Section 3.8.4.II.6. Therefore, the staff finds these certifications to be acceptable.

3.8.4.4.15 Testing and Inservice Surveillance Requirements

Section 3.8.4.7, "Testing and Inservice Inspection Requirements," states that there is no testing or inservice surveillance beyond the quality control tests performed during construction, which is in accordance with ACI 349-13 and AISC N690-18. Further, the applicant issues a COL Item 3.8-2 for a COL applicant that references the NuScale Power Plant US460 will describe the site-specific program for monitoring and maintenance of the Seismic Category I structures in accordance with the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," as discussed in RG 1.160. Monitoring is to include below-grade walls, ground water chemistry, if needed, base settlements, and differential displacements. The staff finds the above-described testing or inservice surveillance and program for monitoring and maintenance to be consistent with DSRS Section 3.8.4.II.7 and therefore acceptable.

Further, Table 1.9-2, shows that the COL applicant is responsible for the water control structures and associated ISI and surveillance programs, in accordance with RG 1.127. The use of RG 1.127 for addressing the site-specific inspection and surveillance programs is consistent with DSRS Section 3.8.4.II.7 and is therefore acceptable.

3.8.4.5 Combined License Information Items

Table 3.8.4-1 lists COL information item numbers and descriptions related to the structural design of Seismic Category I structures, other than containment, from, Table 1.8-2.

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

Item No.	Description	FSAR Section
COL Item 3.8-1	An applicant that references the NuScale Power Plant US460 standard design will provide the design of the reactor flange tool.	3.8.4.1.5.4

COL Item 3.8-2	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific program for monitoring and maintenance of the seismic Category I structures in accordance with the requirements of 10 CFR 50.65 as discussed in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Monitoring is to include below-grade walls, groundwater chemistry if needed, base settlements, and differential displacements.	3.8.4.7

3.8.4.6 Conclusion

The staff finds that the criteria used in the analysis and design of NuScale’s Seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed on each structure during its service lifetime conform with established criteria, codes, standards, and specifications and are therefore acceptable to the NRC staff. On this basis, the staff concludes that the design of NuScale’s Seismic Category I structures other than containment (addressed in SER Section 3.8.2) is acceptable and meets the relevant requirements described in Section 3.8.4.3 of this SER.

3.8.5 Foundations

3.8.5.1 Introduction

This section documents the staff’s review of areas related to the structural design of Seismic Category I foundations for the RXB and CRB. DSRS Section 3.8.5, “Foundations,” provides guidelines and acceptance criteria for reviewing issues related to the foundations of all Seismic Category I structures.

3.8.5.2 Summary of Application

FSAR Sections 3.8.4, 3.8.5 and Appendix 3B, “Design Reports and Critical Section Details,” provide information on the structural design and analysis of the Seismic Category I RXB and CRB structures and foundations.

The applicant described structures and foundations; applicable codes, standards, and specifications; design and analysis procedures; loads and load combinations; structural acceptance criteria; settlement; thermal loads; construction loads; leak detection; materials, quality control, and special construction techniques; and testing and inservice inspection (ISI) requirements. The applicant also described COL information items related to structural design aspects of Seismic Category I structures.

The applicant indicated that the Seismic Category I structures (other than the containment) are portions of the RXB and the CRB. Both buildings designed based upon generic soil profiles and FSAR Section 2.0 enveloping site parameters are site independent. The applicant stated that the CRB is located northwest of the RXB and that there is an underground ductbank between the two buildings. The applicant performed the static and seismic analyses using ANSYS finite element analysis software.

FSAR Appendix 3B, provides a design report for critical sections. In accordance with Appendix 3B, the applicant selected the critical sections based on whether they (1) perform a safety-critical function, (2) are subjected to large stress demands, (3) are considered difficult to design or construct, or (4) are considered to represent the structural design.

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

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

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

3.8.5.3 Regulatory Basis

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

- GDC 1, as it relates to safety-related structures being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- GDC 2, as it relates to the design of the safety-related structures being capable to withstand the most severe natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, and the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.
- GDC 4, as it relates to appropriately protecting safety-related structures against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
- GDC 5, as it relates to not sharing safety-related structures among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.
- 10 CFR Part 50, Appendix B, as it relates to the QA criteria for nuclear power plants.

The guidance in DSRS Sections 3.8.4 and 3.8.5 list the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other DSRS sections. In addition, the following guidance documents provide acceptance criteria that confirm the above requirements have been adequately addressed:

- RG 1.206, as it provides the basis for evaluating the description of structures to be included in a DC or a COL application.
- RG 1.142, as it describes methods and procedures for the analysis, design, construction, testing, and evaluation of safety related nuclear concrete structures (excluding concrete reactor vessels and concrete containments) that comply with NRC regulations.

3.8.5.4 Technical Evaluation

The staff reviewed FSAR Section 3.8.5, in accordance with DSRS Section 3.8.5. DSRS Section 3.8.5 describes acceptance criteria to meet the relevant requirements of the NRC's regulations pertaining to foundations of all Seismic Category I structures. Consistent with DSRS Section 3.8.5, the staff reviewed (1) the description of the foundations, (2) applicable codes, standards, and specifications, (3) loads and load combinations, (4) design and analysis procedures, (5) structural acceptance criteria, (6) materials, quality control, and special construction techniques, and (7) testing and inservice surveillance requirements. The staff also reviewed applicable COL information items.

3.8.5.4.1 Description of Foundations

The staff reviewed the descriptions of the foundations to ensure that they contain sufficient information to define the primary structural aspects and elements that are relied upon to perform the safety-related functions of these structures. The primary function of a foundation is to transmit the loads imposed by the superstructure to the underlying supporting media, rock, or soil. The staff's review also ensures that the foundation design meets the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and that is in accordance with DSRS Acceptance Criterion 3.8.5.II.1.

FSAR, Section 3.8.5.1, "Description of Foundations," describes the physical and functional characteristics of the reinforced concrete basemats of the RXB and CRB for the NuScale US460 Power Plant. The applicant identified the RXB and CRB as Seismic Category I. FSAR Tables 3B-11, 3B-14 and 3B-22 and FSAR Figures 3B-30, 3B-32, and 3B-51 contain the information of steel reinforcement for the RXB and CRB basemats.

The applicant described the RXB basemat dimensions as 70.1 m (230 ft) by 47.2 m (155 ft), with a minimum thickness of 2.44 m (8 ft). The applicant indicated that the foundation top of concrete (TOC) elevation is 7.6 m (25 ft), except for the refueling pool area which has a TOC elevation of approximately 7.9 m (26 ft), **[[**
]]. FSAR Tables 3B-11 and 3B-14 and FSAR Figures 3B-30 and 3B-32 provide the information of steel reinforcement and reinforcement layout for the RXB basemat. Typical longitudinal reinforcement of the RXB basemat consists of four layers of #11 bars centered at 30 cm (12 in) each way on top and bottom surfaces, and typical shear reinforcement consists of #4 ties centered at 30 cm (12 in) each way.

The applicant described the CRB basemat dimensions as 36.6 m (120 ft) by 16.8 m (55 ft), with a thickness of 1.5 m (5 ft). The applicant indicated that the **[[ ]]**. FSAR Table 3B-22 and Figure 3B-51 provide the information of steel reinforcement for the CRB basemat. Typical longitudinal reinforcement of the CRB basemat consists of two layers of #11 bars centered at 30 cm (12 in) each way on top and bottom surfaces, and typical shear reinforcement consists of #3 ties centered at 30 cm (12 in) each way.

The staff reviewed the descriptions of the foundations for RXB and CRB buildings to ensure that they contain sufficient information to define the primary structural aspects and elements that are relied upon to perform the safety-related functions of these structures. The primary function of a foundation is to transmit the loads imposed by the superstructure to the underlying supporting media, rock, or soil. The applicant's description meets the applicable requirements in 10 CFR Part 50, Appendix B, General Design Criterion (GDC) 1 thus in accordance with the guidance in DSRS Acceptance Criterion 3.8.5.II.1.

3.8.5.4.2 Applicable Codes, Standards, and Specifications

FSAR Section 3.8.5.2 refers to FSAR Section 3.8.4 for the codes, standards, and specifications used to design and construct the RXB and CRB structures and foundations. FSAR Section 1.9 presents the regulatory guides applicable to design and construction of the Seismic Category I portions of the RXB and CRB. The applicant indicated that they would use the latest endorsed edition of the ASTM standards at the time of the construction.

Section 3.8.4 of this SER documents the staff conclusion and review of the applicable codes, standards, and specifications used for the structures and foundations.

3.8.5.4.3 Design and Analysis Procedures

DSRS Section 3.8.5 provides review guidance pertaining to the design and analysis procedures of foundations. FSAR Tables 3B-11, 3B-14, 3B-22 and FSAR Figures 3B-30, 3B-32, and 3B-51 provide the steel reinforcement patterns for the RXB and CRB foundations based on the structural analyses and calculations; and FSAR Section 3.8.5.3 "Design and Analysis Procedures" describes the RXB and CRB stability analysis model. FSAR Appendix 3B summarizes the structural design and analysis of the RXB and CRB. The applicant also addressed the capacity of sections, forces and moments at critical locations, and design checks, boundary conditions for each foundation model, soil stiffness conditions, and settlement evaluations. DSRS Section 3.8.5.II.4 provides review guidance on the evaluation of stiff and soft spots in the foundation soil to maximize the bending moments used in the design of mat foundations. In FSAR, Section 3.8.5.3.3 and Table 1.8-1 "Combined License Information Items", the applicant provided COL Item 3.8-3 for an applicant that references the NuScale Power Plant US460 standard design to identify local "stiff and soft spots" in the foundation soil and to address these in the design of foundations, as necessary.

The applicant also employed the ANSYS computational software to generate finite element models (FEMs) simulating the structural response under static and dynamic loads as described in FSAR Sections 3.7 and 3.8 for the design and analysis of the RXB and CRB foundations as appropriate. The foundations were modeled using solidshell (SOLSH190) elements for RXB and shell (SHELL181) elements for CRB, respectively. The soils were modeled using Soil Type 7, 9, and 11, along with Soil Type 7 with soil separation (soil separation case applies to the RXB only) in analysis, as applicable.

Based on the review, the staff determined that the applicant provided an appropriate level of information for the design and analysis procedure used for the Seismic Category I foundations. The staff also determined the use of the ANSYS FEM to design and analysis the RXB and CRB foundations to be acceptable because the ANSYS compute code is widely recognized in the industry and has sufficient history of use to demonstrate its suitability. The staff concludes that the applicant meets the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.3.1 Reactor Building Stability Analysis Model Description

FSAR Section 3.8.5.3.1 provides the uplift, sliding and overturning stability analysis model description for the RXB. The applicant used a static force equilibrium method to develop equations for each of the stability cases to determine the factor of safety (FOS). FSAR Table

3.8.5-2a contains the parameters used for the RXB stability analyses. The applicant took results from the RXB portion of the double building harmonic analysis and post-processed them. From this analysis, the applicant took transfer functions in three directions for each soil type of interest and extracted, interpolated, and convolved with input seismic motions to retrieve a time history of resultant forces and moments at the center of the RXB basemat. Then the applicant used these time histories to form the basis of the demand forces and are compared against resisting forces to establish the FOS.

The applicant indicated that the RXB has a center of gravity more than 20 ft below site grade elevation and that in accordance with ASCE 43-19 for a deeply embedded structure with a center of gravity below the site grade elevation on each perimeter wall, demonstration of sliding and overturing stability is not required. However, the applicant decided to perform a completeness calculation to demonstrate the FOS values are greater than or equal to required 1.1.

Based on the review, the staff finds the applicant’s approach acceptable for the stability analyses model of the RXB. The applicant’s description meets DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.3.2 Control Building Stability Analysis Model Description

FSAR Section 3.8.5.3.2 provides the uplift, sliding and overturning stability analysis model description for the CRB conservatively assumed as a surface-founded structure. FSAR Table 3.8.5-6 contains the parameters used for the CRB stability analyses.

The applicant indicated that it performed a linear elastic analysis using a force equilibrium method similar to the RXB. The applicant further conducted a nonlinear transient analysis because the FOS calculated from the linear elastic analysis of load combination with the seismic demand did not meet an acceptable FOS required for sliding and overturning. The applicant considered a hybrid cracked case in its nonlinear model for load combination with the seismic demand. The applicant indicated that the non-linearity stems from the interface between the CRB base and the underlying soil, which is modeled as a frictional surface that allows both sliding and gap formation. The applicant used the Hilber-Huges-Taylor (HHT) implicit time integration method for the transient analysis and verified the convergence solution by repeating the analysis for selected cases with reduced time steps.

The applicant included COL Item 3.8-3 so an applicant that references the NuScale Power Plant US460 standard design will identify local stiff and soft spots in the foundation soil and address them in the design, as necessary.

Based on the review, the staff finds the applicant’s approach acceptable for the stability analyses model of the CRB. The applicant’s description meets DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.4 Loads and Load Combinations

The staff reviewed loads and load combinations used for the foundations to ensure that they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with the guidance in DSRS Acceptance Criterion 3.8.5.II.3.

FSAR Section 3.8.4 presents information for loads and load combinations used for the design of RXB and CRB, including the design of the foundations.

- FSAR, Section 3.8.4.3.3, “Earth Pressure(H),” describes that the embedded exterior walls of the buildings are subjected to lateral soil pressure loads induced by two types of loads, static soil pressure and soil-structure-interaction dynamic soil pressure. The staff noted that applicant correctly described the lateral soil pressure loads for both static and dynamic cases on embedded structures, including RXB.

- FSAR, Section 3.8.4.3.3, “Earth Pressure(H),” explains that the buoyant force is the upward pressure exerted on the bottom of the foundation during a saturated condition. The staff noted that applicant correctly described the buoyant force as equal to the volume of the building below grade multiplied by the density of water.

- FSAR, Section 3.8.4.3.22, “Other Loads,” describes construction loads, and operation with less than 6 NPMs:

    - FSAR Section 3.8.5.9, “Construction Loads,” describes the construction loads on the basemats of the RXB and CRB. The RXB basemat will be poured in a very short time, and the main loads (the pool water, the NPMs) will be added after RXB construction is completed. The staff do not identify concerns about construction-induced settlement for the RXB and CRB basemats.

    - The applicant performed a study to evaluate the dynamic effects of an earthquake when operating with less than 6 NPMs. FSAR Section 3.7.2.10, Section 3.7.2.10, “Sensitivity Studies on Soil Separation, Empty Dry Dock, and Modularity,” and Section 3.7.2.10.2, “Sensitivity Study Results,” report that the difference in results between operation with 6 NPMs and operation with fewer NPMs in place is small and within the capacity of the building design.

3.8.5.4.4.1 Stability Load Combinations

The applicant considered four load combinations for the assessment of stability for flotation, uplift, sliding, and overturning for RXB and CRB:

A. $D + H + W$
B. $D + H + E_S$
C. $D + H + (W_t \text{ OR } W_h)$
D. $D + F' + E_S$

The applicant defined the dead load of a structure as “D”; the weight and pressure of soils as “H”; the operating basis wind load as "W"; loads effects from SSE as “E_S”; loads generated by the design-basis tornado as “W_t”; loads generated by the design basis hurricane as “W_h”; and the buoyant force as “F’”.

The loads and load combinations used for the design of the RXB and CRB, including the design of the foundations, are discussed in FSAR Section 3.8.4. The applicant indicated that the OBE is established as one-third of the SSE. Therefore, in accordance with DSRS Acceptance

Criterion 3.8.5.II.4. the OBE is not a design-basis ground motion for Seismic Category I structures and no specific analysis is required.

The applicant indicated that for the RXB, load combinations A and C do not need to be analyzed because wind loads in these combinations are bounded by the SSE in combination B. For the CRB, the applicant analyzed for load combination C and load combination D because the hurricane reactions have the highest amplitude. The applicant indicated that it did not consider soil weight and pressure around the basemat because the base of the CRB is shallow (1.5 m (5 ft)).

Thus, the applicant concluded that the load combinations B and D, as described above are bounding for the stability assessment for the RXB and load combinations C and D, as described above, for the CRB structures.

The staff reviewed the load combinations considered by the applicant against the DSRS acceptance criterion 3.8.5.II.3 and concludes that the load combinations used to check against sliding and overturning attributable to earthquakes, winds, tornados hurricanes and against flotation are bounding for the stability assessment. The stability load combinations are acceptable because they are in accordance with DSRS Acceptance Criterion 3.8.5.II.3.

3.8.5.4.4.2 Lateral Soil Force and Seismic Loads

FSAR Section 3.8.5.4.1, “Lateral Soil Force and Seismic Loads,” states that the RXB is an embedded structure; therefore, surrounding soil imposes lateral soil pressures to the embedded structure. The applicant indicated that the CRB is not embedded in the soil, therefore the exterior walls are not subject to static and dynamic lateral soil pressure loads.

FSAR Table 3.8.5-6 provides input evaluation parameters for CRB including static coefficient of friction (CoF) of 0.58 and Kinetic CoF of 0.5 between concrete and underlying soil, which sets the basis on required minimum static CoF and kinetic CoF (Refer to Table 2.0-1, “Site Parameters,” in ML24215A044). FSAR Section 2.5.4 indicates that the friction is defined between the concrete and clean gravel, gravel-sand mixture, or coarse sand with a friction angle of 30 degrees.

FSAR Section 3.8.4.3.3 describes the values of total maximum lateral soil pressure on walls and FSAR Section 3.8.5.4.1 provides the equation to determine the total lateral static effective soil forces on walls.

FSAR Table 3.8.5-2a lists the surcharge load as 12 kPa (250 psf), which is used in the design calculations for the RXB embedded walls subjected to static lateral soil pressure.

FSAR Section 3.8.5.4.1 calculates the lateral soil forces for the RXB. The forces on the RXB walls are calculated as 223,505 kN (50,246 kips) for the north and south walls and 150,127 kN (33,750kips) for the east and west walls. The staff performed an independent check of the calculations and determined that the applicant calculated the forces properly.

Based on the review of the parameters and independent check of the calculations, the staff concludes that the applicant correctly calculated the lateral soil forces and pressure and the seismic base reactions for RXB and CRB. The applicant also met DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.4.3 Effective Vertical Load

FSAR Section 3.8.5.4.2 describes the effective vertical load. The effective vertical load is an important stabilizing force for stability evaluations of the buildings. The applicant calculated the effective dead weights of the RXB and CRB by subtracting the dead weight of the buildings from the buoyancy forces and lists them as 630.8 MN (141,800 kips) and 86.7 MN (19,492 kips), respectively.

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

3.8.5.4.4.4 Friction-Resistant Loads

FSAR Section 3.8.5.4.3 describes the friction-resistant loads. The friction-resistant loads consist of (1) total sliding frictional resistance on the foundation surface from effective vertical load and (2) friction forces resulting from at-rest earth pressures. Frictional resistance loads are considered to stabilize the structure against floating, sliding, and overturning loads since the RXB is a deeply embedded structure.

FSAR Section 3.8.5.4.3.1 describes the passive and active earth pressures and corresponding friction force. The applicant calculated the passive earth pressure coefficient K_p and passive pressure force acting on each wall; and the active earth pressure coefficient K_a and active pressure force acting on each wall.

FSAR Sections 3.8.5.4.3.2 and 3.8.5.4.3.3 describe overturning moment resistance in east-west direction and in north-south direction, respectively. FSAR Figure 3.8.5-4 and Figure 3.8.5-5 provides illustration of RXB for the east-west and north-south overturning moments with their associated moment arms, respectively.

FSAR Section 3.8.5.4.3.4 describes how factors of safety against flotation, sliding, and overturning is derived for RXB. FSAR Figure 3.8.5-1 through Figure 3.8.5-5 provide free body diagrams of the forces at play when establishing each FOS for the equation derived in FSAR Section 3.8.5.4.3.4 for flotation, sliding, or overturning. Several terms related to demand come from the seismic results covered in FSAR Section 3.7.2.

The staff reviewed the information on how to determine friction-resistant loads and the approaches to derive the factors of safety for the RXB. Based on the review, the staff finds the applicant's description in the FSAR acceptable for describing the friction-resistant loads for RXB because (1) adequate consideration has been given to each frictional resistance scenario, including resistance loads for flotation, sliding, or overturning, and (2) the equations of factors of safety against flotation, sliding, and overturning are reasonably established based on resistance capacities and load demands. The applicant's description also met DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.5 Results Compared with Structural Acceptance Criteria

The staff reviewed the structural acceptance criteria used for the foundations to ensure they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.5.

3.8.5.4.5.1 Reactor Building Stability

In FSAR Section 3.8.5.5. the applicant used a static force equilibrium method to determine factor of safety for the RXB against overturning, sliding, and uplift.

The applicant calculated FOS for RXB uplift for a flooding event acting simultaneously with the maximum vertical seismic force. The staff noted that the applicant computed a FOS of 1.25, which complies with the minimum FOS of 1.1 required.

For the RXB sliding stability calculation, the applicant introduced a scaling factor, Ci, related to passive and active pressure from soil. This factor is iterated upon until the minimum factor of safety across all fields investigated is equal to the acceptable value of 1.1. The FSAR Table 3.8.5-13 contains the RXB sliding factors of safety for every seismic/soil configuration. The staff noted that the applicant used a passive and active factor, ci of 0.29, resulting in a FOS of 1.45, which complies with the minimum FOS of 1.1 required.

The applicant calculated the RXB overturning stability at every time step in each seismic event time history and evaluated each edge of the RXB basemat separately. FSAR Table 3.8.5-14 contains the RXB overturning moment factor of safety for every seismic/soil configuration. The staff noted that the applicant's resulting minimum FOS was 1.1 for a soil type with a passive and active pressure factor, Ci of 0.29, which complies with the minimum FOS of 1.1 required.

The staff reviewed FSAR Table 3.8.5-3 which contains a summary of the applicant's calculated factor of safety for the RXB against uplift (flotation), sliding and overturning. The staff noted that all results comply with the required minimum factor of safety of 1.1. Based on the review and comparison of the applicant stability analysis results with the structural acceptance criteria the staff finds that the RXB stability analysis meets applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.5.

3.8.5.4.5.2 Control Building Stability

FSAR Section 3.8.5.5.2 provides the uplift, sliding, and overturning stability evaluation of the CRB. The applicant performed the evaluation using a linear elastic analysis and a nonlinear analysis.

The applicant calculated the CRB FOS for uplift using the resistance force of the building dead weight; and buoyancy and the peak vertical forces of the base reaction calculated from the hurricane and seismic analyses as driving forces. The staff reviewed FSAR Table 3.8.5-15 which contains the CRB uplift calculated FOS and noted that all FOS complies with the minimum FOS of 1.1 required.

The applicant performed a nonlinear transient analysis for the calculation of FOS of sliding because for the stability analysis, performed with the force equilibrium method, the load combination with the seismic demand, do not met the minimum factor of safety of 1.1 against sliding. The CRB sliding FOS time histories are shown in FSAR Figure 3.8.5-16a, Figure 3.8.5-16b, and Figure 3.8.5-16c for seismic events of interest with the Soil Type 7, Soil Type11, and Soil Type 9, respectively. The applicant tabulated the sliding result in FSAR Table 3.8.5-17, where the maximum absolute sliding from the nonlinear transient analysis is 33 mm (1.3 inches). The applicant considered this value to be acceptable given the level of conservatism in the analyses and the distance of the CRB to the nearby SC-II structure. Based on its review, the staff finds the applicant's approach acceptable because the applicant performed detailed nonlinear sliding analyses which would provide more realistic results for CRB. The staff also reviewed the tabulated results in FSAR Table 3.8.5-17 and confirmed that the results would not cause structural damage to CRB structure due to its sliding.

For the CRB overturning stability the applicant indicated that overturning is not a concern under hurricane load. The applicant stated that the overturning stability is further analyzed through nonlinear transient analyses for the load combination with seismic load.

The staff reviewed response to audit question 3.8.5-8 (ML24346A142) and the applicant referenced report EC-103147 "Stability Analysis of the SC-I Category Control Building Structure" where the applicant provided results of the stability analysis of the SC-I structures including the overturning stability analyzed through a non-linear transient analysis for all load combinations. The staff confirmed that the non-linear transient results for overturing stability are negligible as shown in FSAR Table 3.8.5-17 with a maximum non-linear transient result of 0.8 mm (0.03 inches) for vertical displacement.

Based on the review and comparison of the applicant stability analysis results with the structural acceptance criteria the staff finds that the CRB stability analysis meets applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.5.

3.8.5.4.5.3 Average Bearing Pressure Approach

FSAR Section 3.8.5.6, "Bearing Pressure Approach," describes the average bearing pressure results and the bearing pressures along the edges. The applicant calculated the mean RXB bearing pressures from SOLID185 elements forming the soil layers under the basemats and by dividing the sum of nodal forces in vertical direction under the basemats by the corresponding areas. The applicant calculated mean bearing pressures under CRB similarly but using nodes at the solid-structure interface beneath the basemat as the CRB basemat used contact elements to connect the soil and the backfill. The total maximum dynamic bearing pressure (static+ dynamic) values are 33.6 ksf for the RXB and 25.1 ksf for the CRB. The applicant used three rows of elements to define the areas for the calculation of toe pressure. FSAR Tables 3.8.5-4 and 3.8.5-5 list the average bearing and toe pressure values under the RXB and CRB basemats, respectively.

Based on its review, the staff finds the applicant's approach acceptable, since it is appropriately formulated as described in DSRS Acceptance Criterion 3.8.5.II.4.N.

3.8.5.4.5.4 Settlement

FSAR Section 3.8.5.7, "Settlement," describes the foundation settlements, including the approach and results. The applicant used a large-scale ANSYS FEM to determine the effect of foundation differential movements of the RXB and RWB comprising of uncracked and cracked structural members, referred to as hybrid static double building (DB) model. The applicant used Soil Type 11, the soft soil profile from the soil libraries, to maximize the effect of the differential movements and further reduced the stiffness of soil by 50 percent to amplify the effect of differential movements or settlements. The applicant applied the 50 percent reduction in soil stiffness to the areas below the basemats and extended it to the entire free-field soil model. The staff reviewed response to audit questions 3.8.5-9 (ML24215A041) and 3.8.5-10 (ML24215A043). As part of the responses, the applicant referenced various calculation reports that the staff reviewed. During its review of reference EC-112976-0 "Differential Settlement Analysis of the Double Building Model" the staff confirmed that the applicant performed calculations that considered a 50-percent reduction of the soft-soil profile and performed spot checks on the calculations and methodology used. The staff noted that 2.4 m (8.0 ft) thick RXB mat foundation and 1.5 m (5.0 ft) thick CRB mat foundation are modeled with single layer of solid-shell (SOLSH190) elements and of shell (SHELL181) elements, respectively. To address staff audit concerns regarding the modelling adequacy for use of single layer of solid-shell or shell element for basemat, the applicant performed a mesh-density evaluation for the RXB and CRB foundation to test the impact of a multi-layered solid-shell element SOLSH190 in their basemat deformation. The applicant modeled mat foundations from one to four elements throughout the thickness. The staff reviewed reference EC-151256 "RXB and CRB Basemat Element Selection and Convergence Evaluation" and noted that the difference in vertical displacement between the RXB static models was less than one percentage difference for the CRB statics models. The staff noted that the calculated vertical displacement was not significantly affected by adding multi layers of elements, and that the differential settlement result is not expected to be significantly affected, considering that under the same load and soil conditions, the vertical settlement will be greater than or equal to the differential settlement.

The applicant calculated differential settlement using the static load combination of dead, live, hydrostatic, and effective earth pressure. In addition, out of conservative considerations, the applicant ignored buoyancy forces in settlement analysis and defining the load combination as follows:

$$U = D + F + L + H$$

where U is total load, D is dead load, F is the hydrostatic loads that stem from the RXB pool, L is live load, and H is the effective earth pressure with surcharge load excluding hydrostatic loads. Because the CRB is surface founded the load combination used for its analysis is D + L. The applicant modeled the lateral effective earth pressure acting on the RXB.

FSAR Table 2.0-1 provides the selected site parameters appropriate for the design. FSAR Tables 3.8.5-7 and 3.8.5-8 list displacement, differential settlement values, and tilt in inches per 50 ft for a set of nodes selected on the RXB basemat and FSAR Tables 3.8.5-11 and 3.8.5-12 list applicable settlement results for the CRB. The maximum vertical displacement for the RXB and CRB are 37.6 mm (1.48 in) and 19.6 mm (0.77 in), respectively.

Based on its review of the information submitted by the applicant, the staff finds the applicant's approach is acceptable because the staff confirmed that the applicant performed calculations that considered a 50-percent reduction of the soft-soil profile (Soil Type 11) stiffness values to conservatively determine the static demand forces for the RXB and CRB foundation designs and determine the maximum differential settlements within each building basemat. The staff confirmed that all settlement resulting from the analysis are bounded by FSAR Table 2.0-1 which provides the selected site parameters appropriate for the design. The applicant's responses and evaluations also meet DSRS Acceptance Criterion 3.8.5.II.4.

3.8.5.4.5.5 Thermal Loads

The staff reviewed FSAR Section 3.8.5.8 to ensure that it meets the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and is in accordance with DSRS Acceptance Criterion 3.8.5.II.4.B.

In FSAR Section 3.8.5.8 the applicant stated that in design of reinforced concrete members including basemat, the design demands from thermal effects due to accident temperature were not directly included in design load combinations. Instead, the applicant considered thermal effects by calculating the capacity of concrete sections by limiting the "usable" axial and bending strains to the allowable strains reduced by the thermal strains. The applicant indicated that the thermal forces and moments are greatly reduced or completely relieved with the progress of concrete cracking and reinforcement yielding.

Based on its review, the staff finds the applicant's description of thermal loads acceptable, since they are self-relieving because of concrete cracking and reinforcement yielding. Concrete cracks act as release points for the built-up stress, therefore reducing the magnitude of internal forces and moments. If thermal forces cause significant stress, the reinforcement can yield, absorbing some of the stress and reducing the overall forces and moments in the structure. Thus, the application meets DSRS Acceptance Criterion 3.8.5.II.4.B and is therefore acceptable.

3.8.5.4.5.6 Construction Loads

The staff reviewed the construction loads induced by the proposed construction sequence and by the differential settlements of the soil under and to the sides of the structures for the foundations to ensure they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.3.

In FSAR Section 3.8.5.9, "Construction Loads," the applicant stated that the main loads (the reactor pool and the NPMs) will be added after the RXB construction is completed. Therefore, the applicant did not consider construction-induced settlement. Accordingly, the RXB basemat design did not consider the loads induced by construction. The CRB basemat is smaller than the RXB basemat, and the concrete will be poured after the RXB basemat in the construction sequence.

The staff finds that the applicant's description stating that the main loads will be added after the completion of the RXB construction is acceptable. The staff also agrees that any loads induced by the construction sequence will be negligible since the main loads will be added after the completion of the RXB construction. Similarly, loads induced by the construction sequence will be negligible in the design of the CRB basemat because it is smaller than the RXB basemat, and the loads will be added after the completion of the CRB construction. Therefore, the staff finds the applicant's conclusions acceptable since the main loads will be added after the completion of RXB and CRB construction, and thus the effects of construction loads are not a concern, which meets DSRS Acceptance Criterion 3.8.5.II.4.M.

3.8.5.4.5.7 Leak Detection

The staff reviewed the design details that prevents and monitor potential leakage from the pool and potential leakage into the RXB from ground water to ensure that they meet the applicable requirements in GDC 1, 2, 4, and 5; and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.4.O.

FSAR Section 3.8.5.10, "Leak Detection," describes the leak detection of pool and ground water into the RXB walls and foundation. Ground water has the potential to leak through the RXB exterior walls through microscopic concrete cracks at a very slow rate of less than 3.8 liters (1 gallon) per day. The applicant concluded that this leak would not be enough to cause an interior flood in any of the rooms that share an exterior wall. However, the plant's concrete maintenance specifications and dewatering system surrounding the RXB would effectively reduce ground water leakage.

FSAR Section 3.8.5.10, states, "A leak chase system is provided in the RXB basemat to detect any leakage from the reactor pool." FSAR Section 9.1.3. describes the pool leakage detection system and SER Section 9.1.3 provides the staff evaluation on the pool leakage detection system.

3.8.5.4.6 Materials, Quality Control, and Special Construction Techniques

The staff reviewed the material, quality control, and special construction techniques used for the foundations to ensure that they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with the guidance in DSRS Acceptance Criterion 3.8.5.II.6.

In FSAR Section 3.8.4.6, the applicant describes the materials, quality control, and special construction techniques for the RXB and CRB, including the foundations. The staff reviewed the material, quality control, and special construction techniques in FSAR, Section 3.8.4.6, regarding their application to the RXB and CRB foundations. FSAR, Section 3.8.4.6, describes the principal construction materials for Seismic Category I structures as concrete, reinforcing steel, structural steel, stainless steel, bolts, anchor bolts, and weld electrodes. FSAR Table 3.8.4-3, provides the material properties for materials considered for structural design and indicates that the minimum typical compressive strength of concrete is 34 MPa (5,000 psi) and 48.3 MPa (7,000psi) for the RXB roof slab and floor slabs. FSAR, Section 3.8.4.6.1.1, also states that the concrete ingredients are cement, aggregates, admixtures, and water. FSAR Sections 3.8.4.6 and 3.8.4.6.1.1 provide the applicable industrial codes and standards and RGs that the materials and quality control shall satisfy, and they specifically refer to ACI 349, ACI 301, and RG 1.142 for the design of Seismic Category I structures.

FSAR Section 3.8.4.6.1.2, states that the steel reinforcing bar material conforms to A615 Grade 60 or A706, Grade 60.

The staff finds the use of these 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 FSAR Section 3.8.5.6, to be acceptable.

3.8.5.4.7 Testing and Inservice Inspection Requirements

The staff reviewed the testing and inservice surveillance requirements used for the foundations to ensure that they meet the applicable requirements in GDC 1, 2, 4, and 5 and 10 CFR Part 50, Appendix B, and are in accordance with DSRS Acceptance Criterion 3.8.5.II.7.

FSAR Section 3.8.5.12 refers to FSAR Section 3.8.4.7, for a description of the testing and inservice inspection requirements for the RXB and CRB foundations. The applicant stated that there is no testing or inservice surveillance beyond the quality control tests performed during construction, which is in accordance with ACI 349, and AISC N690. In FSAR Section 3.8.4-7 the applicant included COL Item 3.8-2, which states that an applicant that references the NuScale Power Plant US460 standard design will describe the site-specific program for monitoring and maintenance of the Seismic Category I structures in accordance with 10 CFR 50.65 as discussed in RG 1.160, where monitoring is to include below-grade walls; ground water chemistry, if needed; base settlements; and differential displacements.

The staff reviewed FSAR, Sections 3.8.5-12 and 3.8.4.7, and concludes that the testing and in service surveillance requirements used for foundations are in accordance with 10 CFR 50.65 and RG 1.160, as addressed in DSRS Section 3.8.5.

3.8.5.5 Combined License Information Items

SER Table 3.8.5-1 lists COL information item numbers and descriptions related to the structural design of Seismic Category I foundations for RXB and CRB.

Table 3.8.5-1: NuScale COL Information Item for Section 3.8.5

Item No.	Description	FSAR Section
COL Item 3.8-2	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific program for monitoring and maintenance of the seismic Category I structures in accordance with the requirements of 10 CFR 50.65 as discussed in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Monitoring is to include below grade walls, groundwater chemistry if needed, base settlements, and differential displacements.	3.8.4.7
COL Item 3.8-3	An applicant that references the NuScale Power Plant US460 standard design will identify local stiff and soft spots in the foundation soil and address these in the design, as necessary.	3.8.5.3.3

3.8.5.6 Conclusion

The staff concludes that the NuScale Power Plant US460 standard design's RXB and CRB foundations is acceptable and meets the regulatory requirements described in Section 3.8.5.3 of this SER.

3.9 Mechanical Systems and Components

3.9.2 Dynamic Testing and Analysis of Systems, Structures, and Components

3.9.2.1 Introduction

This section of the SER evaluates the analytical methodologies, testing procedures, and dynamic analyses used by the applicant to ensure the structural and functional integrity of the piping systems, mechanical equipment, RVIs including the internal SG, and their supports under vibratory loadings, including those caused by fluid flow, short term transients, and postulated seismic events.

This section addresses six main areas of review:

(1) Piping vibration, thermal expansion, and dynamic effects testing (3.9.2.4.1)

(2) Seismic analysis and qualification of Seismic Category I mechanical equipment (3.9.2.4.2)

(3) Dynamic response analysis for RVIs and SGs under operational flow transients and steady state conditions (3.9.2.4.3)

   - Design and Operation Summary
   - Analytic Flow-Induced Vibration Evaluation
   - Forcing Function Methodologies and Assumed Flow Velocities
   - Structural Mode Shapes and Resonance Frequencies
   - Structural Damping
   - Turbulent Buffeting (TB) Analysis
   - Flutter and Galloping Susceptibility
   - Vortex Shedding (VS) and Fluid-Elastic Instability (FEI) Susceptibility
   - Acoustic Resonance (AR) Susceptibility
   - Leakage Flow Instability (LFI) Susceptibility
   - Density Wave Oscillation
   - Benchmarking Testing

(4) Preoperational flow-induced vibration testing of RVIs and SGs (3.9.2.4.4)

   - Flow Induced Vibration Testing of SGs in Test Facility 3
   - Initial Startup Testing of NPM
   - Inspections

(5) Dynamic system analysis of the RVIs and SGs under faulted (service level D) conditions (3.9.2.4.5)

   - Seismic Analysis
   - Short-Term Transient Analysis
   - Stress Evaluation of RVIs and SGs

(6) Correlations of RVIs and SG vibration tests with analytical results (3.9.2.4.6)

3.9.2.2 Summary of Application

FSAR Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment,” presents criteria, testing, and dynamic analyses employed to ensure structural and functional integrity of piping systems, mechanical equipment, and reactor internals and their supports under dynamic and vibratory loading, including those due to fluid flow during normal plant operation, transient conditions, and postulated seismic events. The NuScale NPM includes an internal SG system which is also evaluated for structural and functional integrity.

FSAR Section 3.9.2.1, “Piping Vibration, Thermal Expansion, and Dynamic Effects” addresses the initial startup testing that is performed to verify that the vibrations and thermal expansion and contraction of the as-built piping systems are bounded by the design requirements.

TR-121353, Revision 2, “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report” (Comprehensive Vibration Assessment Program (CVAP) analysis technical report), issued January 2025 (ML25023A215 (proprietary) and ML25023A214 (non-proprietary)), is referenced in FSAR Section 3.9.2.3, “Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady State Conditions,” and Section 3.9.2.4, “Flow-Induced Vibration Testing of Reactor Internals Before NuScale Power Module Operation.” In addition, the applicant has submitted technical report TR-121354, Revision 1, “NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report,” (MIP technical report), issued August 2024 (ML24222A529 (proprietary) and ML24222A528 (non-proprietary)). FSAR Section 14.2 describes the SG prototype testing (Test #65) and the NPM initial startup vibration testing (Test #102).

The CVAP technical report describes the screening procedures and provides the results of the flow-induced vibration (FIV) analyses of (1) RVIs and structures, (2) SG components, and (3) primary and secondary RCS piping, up to the NPM disconnect flange. Components with small margins of safety against FIV effects were identified for validation testing. The MIP technical report describes a SG mockup, called the Società Informazioni Esperienze Termoidrauliche (SIET) test facility-3 (TF-3), which was built and tested in accordance with TR-121354, Revision 1. The TF-3 was tested over a wide range of flow conditions to confirm that significant SG FIV caused by VS and FEI will not occur in the NPM. The initial startup testing for FIV effects on the prototype NPM will include a set of external pressure sensors to detect any unexpectedly strong FIV of the RVIs and SGs. Prototype NPM initial startup testing will also confirm there are no strong ARs in the containment system (CNTS) steam piping.

FSAR Section 3.9.2.5, “Dynamic System Analysis of the Reactor Internals under Service Level D Conditions,” and Appendix 3A, “Dynamic Structural Analysis of the NuScale Power Module,” describe the structural and dynamic analyses of the NPM. Dynamic analyses for ASME Service Level D events include SSE and blowdowns induced by pipe ruptures and inadvertent valve actuations. Appendix 3A, references Technical Reports TR-121515, Revision 1, “US460 NuScale Power Module Seismic Analysis,” (seismic analyses technical report) issued November 2024 (ML24327A037 (proprietary) and ML24327A036 (non-proprietary)) and TR-121517, Revision 1, “NuScale Power Module Short-Term Transient Analysis,” (short term transient analyses technical report) issued August 2024 (ML24243A010 (proprietary) and ML24243A009 (non-proprietary)), for additional details. The seismic analyses were performed in three phases. First, a finite element model of the RXB, including linearized models of the NPMs, was analyzed with ANSYS for several postulated earthquake loading time histories and several soil types. A nonlinear model of a single NPM was then analyzed using bounding time histories of accelerations at the support locations. Short term transient loads induced by blowdown events were simulated with NRELAP5 and ANSYS. NRELAP5 was used to compute boundary conditions from thermal hydraulic analyses which were applied to the ANSYS model to compute accelerations and loads. The calculated in-structure time histories from the seismic and short term transient NPM analyses were saved along with in-structure response spectra from the seismic analyses for subsequent ASME Service Level D stress analyses of RVIs and SG tubes and supports.

FSAR Section 3.9.6, "Correlations of Reactor Internals Vibration Tests with the Analytical Results," only states that future testing results will be compared to previous analysis results. Any significant deviations would require re-analyses and reconciliation with the test results.

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

Technical Reports:

- TR-121515, Revision 1, "US460 NuScale Power Module Seismic Analysis"
- TR-121517, Revision 1, "NuScale Power Module Short-Term Transient Analysis"
- TR-121353, Revision 2, "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report"
- TR-121354, Revision 1, "NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report"

3.9.2.3 Regulatory Basis

The following relevant NRC regulatory requirements apply to this review:

- GDC 1, as it relates to the design, fabrication, erection, and testing of SSCs in accordance with the quality standards that are commensurate with the importance of the safety function to be performed
- GDC 2, as it relates to the ability of SSCs, without loss of capability to perform their safety functions, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads, and to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics
- GDC 4, as it relates to the protection of SSCs against dynamic effects, including those of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit
- GDC 14, as it relates to designing SSCs of the RCPB to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.
- GDC 15, as it relates to designing the RCS with sufficient margin to assure that the RCPB is not exceeded during normal operating conditions, including AOOs

- Appendix B to 10 CFR Part 50, as it relates to the QA criteria for the dynamic testing and analysis of SSCs

- Appendix S to 10 CFR Part 50, as it relates to certain SSCs that must be designed to remain functional for an SSE

- 10 CFR Part 50.55a, as it relates to the design, fabrication, erection, and testing of SSCs in accordance with the quality standards that are commensurate with the importance of the safety function to be performed

SRP Section 3.9.2 lists the acceptance criteria adequate to meet the above requirements and review interfaces with other SRP sections. In addition, the following guidance documents provide general acceptance criteria that confirm that the above requirements have been adequately addressed:

- RG 1.20, Revision 4, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," issued July 2015, as it relates to the vibration analysis and testing methodologies of the RVIs

- RG 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," issued March 2007, as it relates to the damping values used for a dynamic analysis

- ASME OM-S/G-2000, "Standards and Guides for Operation of Nuclear Power Plants" (ASME Operation and Maintenance of Nuclear Power Plants Code Standards and Guides (OM Code), 2000 Edition), Part 3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems," and Part 7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems," as they relate to guidance for test specifications, as endorsed by SRP 3.9.2

3.9.2.4 Technical Evaluation

3.9.2.4.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

FSAR Section 3.9.2.1, addresses the initial startup testing that is performed to verify that the vibrations and thermal expansion and contraction of the as-built piping systems are bounded by the design requirements. The piping systems in the initial startup testing program include (1) ASME BPV Code, Section III, Class 1, 2, and 3 piping systems, (2) high-energy piping systems inside Seismic Category I structures or those whose failure would reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and (3) Seismic Category I portions of moderate-energy piping systems located outside of the containment.

FSAR Section 3.9.2.1, states that the vibration, thermal expansion, and dynamic effect elements of this test program are performed during preoperational testing and initial startup testing. The preoperational tests are performed to demonstrate that the piping system components meet functional design requirements and that piping dynamic effects are acceptable. If test acceptance criteria are not met, corrective actions (e.g., reanalyzing with as-built values) are implemented, and the systems are retested. The initial startup testing is performed after the reactor core is loaded into a reactor module. These tests determine that the vibration level and piping reactions to transient conditions are acceptable and are bounded by the analyses. If the vibration levels are not bounded, the evaluations use the vibration level from the testing as input to verify that the design is acceptable. FSAR Section 3.9.2.1 lists the initial startup tests that included in FSAR Section 14.2 to verify the piping systems are within the thermal expansion and vibration limits.

FSAR Section 3.9.2.1.1, “Piping Vibration Details,” states preoperational tests and initial startup tests demonstrate that piping systems withstand vibrations resulting from normal operation, including anticipated operational occurrences. If excessive vibration is observed that is outside the bounds of the analyses, a re-analysis to determine the cause and to identify the corrective action is performed. Vibration test specifications are developed in accordance with ASME Operation and Maintenance of Nuclear Power Plants, Division 2, 2017 Edition, Part 3. SRP Section 3.9.2, Revision 4, references the ASME OM Standards and Guides 2012 Edition. The NRC staff finds that the use of ASME OM Code, Division 2, 2017 Edition, is acceptable because the provisions for piping vibration and thermal expansion testing are equivalent.

FSAR Section 3.9.2.1, includes COL Item 3.9-2, for the COL applicant to complete an assessment of piping systems inside the RXB to determine the portions of piping to be tested for vibration, thermal expansion, and dynamic effects. The COL applicant may select piping systems for the vibration testing using the piping vibration screening and analysis results of the CVAP. The staff finds that the COL item adequately addresses the assessment and selection of the piping system for vibration, thermal expansion, and dynamic effect testing during initial startup testing. Additionally, ASME OM Code, Division 2, 2017 Edition, Part 3, does not specify the criteria for selecting piping for vibration testing; therefore, considering the screening and analysis results of the CVAP for the selection of piping systems for vibration testing is an acceptable approach.

FSAR Section 3.9.2.1.1.1, “Main Steam Line Branch Piping Acoustic Resonance,” addresses the concern of potential vibration or fatigue failure of main steam line branch piping due to flow-excited ARs. COL Item 3.9-3 addresses the detailed design of the main steam piping by the COL applicant, ensuring the detailed design of the MS line considers the phenomenon of AR and the piping vibration screening and analysis results of the CVAP. The staff finds that the COL item and the process used to complete the detailed design of the MS line to avoid AR is acceptable because COL Item 3.9-3 will ensure that the design of the piping systems will preclude significant ARs at pipe branches.

FSAR Section 3.9.2.1.2, “Piping Thermal Expansion Details,” states that the thermal expansion testing verifies that the design of the piping systems tested prevents constrained thermal contraction and expansion during normal operation. In addition, the tests verify that the component supports can accommodate the expansion of the piping during normal operation. FSAR Section 14.2, describes selected planned piping thermal expansion measurement tests. Test specifications for thermal expansion testing of piping systems during preoperational and startup testing will be made in accordance with ASME OM Code, Division 3, 2017 Edition, Part 7. The staff finds that performing the piping thermal expansion testing according to OM Code, Division 3, 2017 Edition, Part 7, is acceptable because this meets the SRP guidance. The initial startup testing provides adequate assurance that the piping and piping restraints of the tested systems can expand without obstruction and within design limits and therefore can withstand thermal effects during normal and transient operating conditions.

3.9.2.4.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

FSAR Section 3.9.2.2, “Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment,” references FSAR Section 3.7; Section 3.10, “Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment”; and Section 3.12. The corresponding sections of this SER include the review of these FSAR sections.

3.9.2.4.3 Dynamic Response Analysis of Reactor Vessel Internals and Steam Generators under Operational Flow Transients and Steady State Conditions

3.9.2.4.3.1 Design and Operation Summary

FSAR Chapter 1 describes the overall plant design, including the NPM, and Section 3.9.5, “Reactor Vessel Internals,” describes the RVI components. Aspects of the design that are relevant to FIV are summarized here. The NPM comprises a reactor core, pressurizer, and two integral once-through helical coil SGs within a cylindrical RPV, which is housed in a cylindrical steel CNV. The NPM operates with natural circulation primary coolant flow, which is much slower than in existing PWRs, reducing the strength of flow-induced forces compared to those in a typical PWR. The NPM rests in a reactor pool of water that acts as a heat sink and allows for passive operation (i.e., pumps are not used to circulate or inject coolant) and passive safety systems (i.e., DHRS and ECCS). Since the NPM has no pumps, there are no pump dynamic forces or inlet flow jets that impinge on reactor components. A power plant comprises up to a maximum of six NPMs.

The NuScale RVI is a first-of-its-kind design and is, therefore, classified as a prototype in accordance with RG 1.20. Following the NuScale RVI qualification as a valid prototype, future NPMs will be considered limited prototype or non-prototypes per RG 1.20. A single NPM is smaller than currently operating PWRs, and outputs power up to 250 megawatts thermal. Unlike traditional PWRs with forced primary coolant circulation, the core flow rate is proportional to the plant’s power. The SGs are integral to the NPM and therefore are evaluated for FIV along with RVIs. Also, all piping systems and valves, including those outside the CNV and up to the NPM disconnect flange, are evaluated for FIV effects. FIV effects on the RVIs, SG, and piping systems and valves are evaluated during normal operation, decay heat removal, and emergency core cooling conditions. Although the RVIs will experience worst case FIV loads during normal operation, the main steam lines and isolation valves may experience stronger FIV loads during ECCS or DHRS operation.

The RPV is mounted within the steel CNV, which operates in a large pool of water. The CNV also contains auxiliary piping, including the chemical and volume control system (CVCS) and piping connection to the DHRS. The RPV is constructed of three sections: the head, upper, and lower sections, with the head welded to the upper section and a flanged bolted connection between the upper and lower sections. Several small RPV upper head penetrations accommodate the pressurizer spray, reactor vent valves (RVVs), reactor safety valves (RSVs), and in-core instrumentation. The CRDMs are mounted on top of the RPV with rods extending downward into the RPV.

The RVIs comprise a core support assembly (CSA) and hot-leg riser system. A lower riser assembly (LRA) rests on the CSA. The upper riser assembly (URA) is suspended from the upper riser hanger plate by the control rod drive (CRD) shaft sleeves and the bottom rests on the lower riser along a mating section and is secured against relative radial motion with pins. A bellows is included in the URA near the URA/LRA interface to allow for small relative movement between the upper and lower risers (primarily caused by thermal expansion) and to minimize the likelihood of significant leakage flow between the hot (inner) and cold (outer) legs.

The dome of the RPV houses the pressurizer system. The primary coolant turns downward below the pressurizer baffle plate and passes the SG tubes in the outer annulus of the RPV. Pressure is regulated by a pair of heater bundles, which may be activated to increase pressure, and two spray nozzles connected to the CVCS, which provide subcooled water at the top of the pressurizer to reduce pressure. The nozzle flow rates are very low and do not generate significant flow-induced forces.

The upper and lower risers are welded assemblies. Internal circular support frames (CRD shaft supports) are attached to the risers to accommodate CRD shafts and in-core instrumentation guide tubes (ICIGTs), which are inserted into the top of the RPV and extend downward through the CRD shaft supports and into the core to monitor and control the reactor. The CRD shafts can move upward and downward, whereas the ICIGTs are stationary. Nominal clearances are specified between the hole boundaries in the shaft supports and the CRD and ICIGT structures.

The once-through SGs consist of two independent bundles of tubes within the annulus between the riser and the wall of the RPV. The tubes for each bundle are welded to tubesheets at two integral feed (about halfway along the RPV) and steam (near the top) plenums, thus forming a pressure boundary between the primary and secondary coolant. The tubes are held in place by arrays of tube support assemblies mounted on upper SG supports attached to the pressurizer baffle plate and interfaced with lower SG supports that are attached to the RPV. A series of set screws preload the inner most tube support hanging back strip against the upper riser shell. Nominal small clearances are specified between the tube supports and tubes, but during operation the tubes are expected to have tight contact with the supports due to thermal and hydraulic forces along with the set screw preloading.

A CVCS purifies the primary coolant as needed. CVCS injection piping protrudes through the RPV wall, passes through the downcomer and terminates in the URA above the core exit.

The reactor operates passively, with primary coolant flowing upward through the core and the lower and upper riser assemblies, then moving radially outward below the pressurizer baffle plate and then downward though the annulus between the riser and RPV wall over the SG tube array. After passing over the SG tubes and through the downcomer, the flow moves radially inward before proceeding upward through the core and riser assemblies again. The secondary coolant enters the bottom of the SG tubes, as preheated subcooled liquid, and travels upward opposite the primary coolant flow direction. As heat is transferred from the primary to the secondary coolant, the secondary coolant within the SG tubes boils and transitions to superheated steam, which then exits into a plenum and steam supply nozzles near the top of the RPV where it travels to the steam turbines. Because the primary flow is passive, the velocity is about 5 to 20 times slower than the flow in a traditional PWR. However, due to the prototype design and relatively smaller size of NuScale internal components compared to traditional PWRs, FIV still needs to be assessed.

The SG tubes also function in conjunction with the DHRS. The DHRS provides secondary side reactor cooling for non LOCAs when normal FW is not available. For DHRS operation, the FW and MS isolation and bypass valves are closed, and the DHRS valves are opened. Water/steam in the secondary loop circulates naturally through the DHRS in the reactor pool and SG loops inside the RPV. The DHRS condensers are connected to the two SG loops, rejecting heat to the water in the reactor pool. During DHRS operation the flow rates through the SG are lower than those during normal operation; therefore, DHRS operating conditions do not need SG FIV evaluation. However, flow over cavities and standpipes in the DHRS piping is evaluated for AR.

The ECCS provides primary side cooling and coolant inventory control for LOCAs. For ECCS operation, two sets of emergency core cooling valves are opened. The RVVs release the primary coolant in the RPV to the CNV, where it condenses on the inner walls. The reactor recirculation valves (RRVs) located above the core also open to allow natural circulation between the condensed water in the annulus, between the CNV and the RPV, and the water within the RPV. Because flow rates throughout the steady-state ECCS conditions are low, FIV loads are small. Also, because the duration of any initial transients is short, any induced alternating stresses do not occur for a significant number of cycles.

Inlet flow restrictors (IFRs) are tightly fitted into all SG tube inlets to add stability against possible density wave oscillation (DWO) behavior in the secondary coolant system. It is well known that DWO can occur in parallel SG channels/tubes filled with fluids at different states (see for example Oh, S., Kim, D., and Lee, J., “Prediction of Density Wave Oscillation in Helical Steam Generators Using the MARS-KS Code,” International Journal of Heat Mass Transfer, Volume 235, 2024, 126226 along with its many references to other papers). NPM-20 SG tubes are filled with subcooled liquid at their inlets, followed by a boiling boundary and a section of two-phase flow (steam and liquid), followed by the dryout location and a final column of superheated vapor at the outlet. There are no restrictors at the tube outlets. Small inlet mass flow oscillations can induce density waves in the two-phase region (which has a varying density throughout) which can sometimes generate a pressure drop in the vapor region that is out of phase with the inlet flow oscillations, reinforcing them. The mechanism is described by Oh, as well as Reyes, “A Semi-Empirical Correlation for the Onset of Density Wave Oscillations in a Helical Coil Steam Generator,” Nuclear Technology, Volume 210, Issue 5, 906-918.

DWO has a characteristic time cycle, usually on the order of tens of seconds. During DWO, flow in some tubes reverses while flow in neighboring tubes flows forward – this is called “incoherent DWO.” In severe cases with high amplitude oscillations the boiling water boundary can approach the inlet, potentially leading to cavitation as well as condensation-induced water hammer (CIWH) in the tube inlet region and in the FW plenum. In very rare cases the oscillations in all tubes are in-phase (all in reverse or all forward at any given instant in time) – this is “coherent DWO.” Incoherent DWO is usually not evident in any of the usually monitored parameters (temperature, pressure), but coherent DWO will induce observable changes, allowing operators to take action to mitigate it.

The NPM-20 IFRs are long rods with narrow circular center orifices which induce a large inlet pressure drop. Any pressure oscillation in a SG tube which pushes the subcooled liquid back toward the inlet will be resisted by the IFR pressure drop, stabilizing the system against DWO initiation. The IFRs also mitigate the strength/amplitude of DWO oscillations should they occur. NuScale acknowledges in FSAR Section 3.9.1.1.1 and in Section 4.2-21 (Transient A21 – Density Wave Oscillations) of ER-101144, “Pressure and Thermal Transient Definitions for Analysis of NSSS Components,” Revision 3 (referenced in response to Audit Question A-5.4.1.3-3, item 16 (ML25013A243 (proprietary) and ML25013A242 (nonproprietary)) that DWO instabilities may occur in the SG at limited transient conditions and when the DHRS system is activated and the FW temperature can no longer be controlled. Following the initial SDAA submission NuScale submitted updated secondary coolant system operating conditions in Section 2.1 (Inputs) of EC-110662, “Primary and Secondary Steady State Parameters,” Revision 2, along with a temperature-based approach for monitoring NPM operating time in DWO conditions in FSAR Section 5.4.1.3 with details in EC-174500, “DWO Approach Temperature Limit,” Revision 0 in response to SDAA Audit Question DWO-SC-25 (ML25013A222 (proprietary) and ML25013A221 (nonproprietary)). NuScale also submitted a series of technical evaluations of the potential FIV-induced effects of operating at severe DWO conditions in response to Audit Questions A-3.9.2-26 (ML24346A148 (proprietary) and ML24346A147 (non-proprietary)) and A-3.9.2-34 (ML24346A158 (proprietary) and ML24346A157 (non-proprietary)). Finally, NuScale updated the SG Technical Specification, Section 5.5.4, “Steam Generator (SG) Program” to require more frequent inspections to ensure tube integrity.

3.9.2.4.3.2 Analytic Flow-Induced Vibration Evaluation

The NRC staff based this evaluation of the applicant’s FIV, RVI, and SG analyses on (1) the CVAP analysis and MIP technical reports and (2) an audit of the applicant’s internal documents, drawings, and test data conducted from March 27, 2023, through August 31, 2024 (ML24211A089). The staff used the audits to assess the details of the analyses. The SER presents only significant aspects of the CVAP analysis and MIP technical reports and audits.

The applicant screened the following components for FIV:

- RVIs
- SG components
- primary and secondary coolant piping up to the NPM disconnect flanges

Based on the screenings, the applicant identified selected components for more detailed FIV evaluations. The staff finds the screening procedures to be acceptable because they are consistent with the guidance in ASME BPV Code, Appendix N, “Dynamic Analysis Methods,” and the open literature.

The applicant evaluated the following components in more detail for FIV effects resulting from primary coolant flow:

- SGs
  - tube support bars
  - hanging backing strip and set screws
  - SG tube support spacer
  - lower SG support
- URA
  - upper riser shells and transition shell
  - upper riser bellows and bellows threaded limit rods
  - set screw assemblies
  - ICIGTs and riser level sensor GTs
  - CRD shaft
  - CRD shaft support
  - CRD shaft sleeve

- LRA
  - lower riser section
  - control rod assembly guide tubes (CRAGTs)
  - CRAGT support plate
  - ICIGT funnels and lower riser ICIGTs
  - upper core plate

- CSA
  - core barrel
  - upper support block assembly
  - CSA mounting brackets
  - reflector block
  - lower core plate
  - fuel pin interface

- Other RVIs
  - RCS injection RVI
  - pressurizer spray RVI
  - thermowells
  - component and instrument ports
  - ECCS valves

- Primary coolant piping
  - RCS injection line tee location
  - CNTS drain valve tee locations

16 small slots in the upper riser permit flow between the upper riser and SG primary coolant regions during DHRS conditions when the top of the riser is uncovered. As discussed later in this SE, the applicant evaluated the potential impacts of the riser holes on FIV.

The applicant evaluated the following components for FIV effects resulting from secondary coolant forward flow:

- Steam generator system (SGS) and CNTS steam piping, MS isolation valves (MSIVs)
- SG steam plenum
- DHRS steam and condensate piping
- SG tubes
- SG tube IFRs
- SGS pressure relief valve branch, CNTS FW drain valve branch

The NRC staff does not usually review SGs as part of an RVI CVAP. However, because the SG tubes are integral to the NuScale reactor module, the staff reviewed it for FIV. The staff finds that the components evaluated for FIV are reasonable and that it is unlikely that any other RVI are susceptible to FIV based on low-flow conditions or robust structural designs, or both.

The applicant addressed the following FIV mechanisms:

- TB – random flow turbulence driving structures into broad-band, usually small amplitude, vibration
- F/G – bluff bodies, like circular or rectangular cross sections, locking into cross-flow into large amplitude vibrations
- VS – vortices shed from the aft ends of objects, usually pipes, which lock-in to pipe structural modes of vibration, leading to large amplitude vibrations
- FEI – multiple adjacent tubes vibrating in various patterns in response to cross flow, leading to very large amplitude vibrations
- AR – flow over side branch openings locking in to acoustic modes in the side branch, leading to high amplitude acoustic pulsations which can in turn lead to strong vibrations
- LFI – flow through narrow gaps which generates oscillating forces which lock on to structural modes of vibration

These are the usual FIV mechanisms evaluated in a CVAP, and the NRC staff finds them to be acceptable. For TB, the applicant also assessed fatigue life and wear associated with intermittent contact and relative motion between adjacent components. All other FIV mechanisms are evaluated only for their potential to occur since they are associated with the "lock in" of structural or acoustic motion with a flow-induced excitation mechanism or instability. Below a so-called "critical flow velocity," determined for each component, this lock-in cannot occur and the flow-induced forces are small. If, however, lock-in were to occur, rapid failure (days or weeks) of the associated SSC would be expected.

For much of its screening and analyses, the applicant relied heavily on the following references:

- ASME BPV Code, Section III, Nonmandatory Appendix N1300, "Flow-Induced Vibration of Tubes and Tube Banks";
- a workbook by M.K. AuYang, "Flow-Induced Vibration of Power and Process on Plant Components: A Practical Workbook," issued 2001;
- a book by R.D. Blevins, "Flow Induced Vibration," 2nd Edition, issued 1990;
- NUREG/CR-6031, "Cavitation Guide for Control Valves," by J.P. Tullis, issued 1993.
- a paper by S.S. Chen on FEI and VS in helical coil SG tubing (see S.S. Chen, "Tube Vibration in a Half-Scale Sector Model of a Helical Steam Generator," *Journal of Sound and Vibration* 91(4), pages 539–569, issued 1983); and
- four papers on LFI by F. Inada referenced in TR-121353

    (1) Inada, F., "A Study on Leakage Flow Induced Vibration From Engineering Viewpoint," PVP2015-45944, ASME 2015 Pressure Vessels and Piping Conference, Volume 4: Fluid-Structure Interaction, July 19–23, 2015, American Society of Mechanical Engineers, New York, NY, 2015,

(2) Inada, F. and S. Hayama, “A Study on Leakage-Flow-Induced Vibrations. Part 1: Fluid-Dynamic Forces and Moments Acting on the Walls of a Narrow Tapered Passage,” *Journal of Fluids and Structures*, issued in 1990: pages 4:395-412,

(3) Inada, F. and S. Hayama, “A Study on Leakage-Flow-Induced Vibrations. Part 2: Stability Analysis and Experiments for Two-Degree-Of-Freedom Systems Combining Translational and Rotational Motions,” *Journal of Fluids and Structures*, issued in 1990: pages 4:413-428, and

(4) Inada, F., “A Parameter Study of Leakage-Flow-Induced Vibrations,” Proceedings of the ASME 2009 Pressure Vessels and Piping Division Conference, July 26–30, 2009, American Society of Mechanical Engineers, New York, NY, issued in 2009.

Unlike previous applicants that have submitted a comprehensive scale model or full-scale plant test data, operating history of a similar design, or all of the above, NuScale has performed less extensive benchmarking to date to substantiate its analysis procedures. The applicant evaluated each FIV mechanism for a given component using a combination of the following:

- forcing function methodologies (from the ASME BPV Code or Au-Yang’s workbook)
- assumed flow velocities (from computational fluid dynamics (CFD) or bulk flow estimates)
- structural cross sections and lengths, mode shapes, and lowest resonance frequencies (from ANSYS FEAs), used to estimate critical velocities for each FIV mechanism
- assumed structural damping

The flow velocities, along with the forcing function methodologies, are required to estimate the flow-induced forces. The faster the flow, the higher the forces. The structural dimensions, boundary conditions, and material properties dictate the shapes and resonance frequencies of modes of vibration, and therefore the structural response functions. FIV from TB is computed by multiplying the estimated flow-induced forces by the structural response functions.

FIV mechanisms that involve flow instabilities and possible lock-in with acoustic or structural resonances are evaluated using criteria in the ASME BPV Code and the Inada references. These criteria generally combine the coincidence of flow-induced forcing and structural or acoustic response frequencies and the structural or acoustic damping. If the force and response frequencies coincide and damping is small, then lock-in and strong vibration or sound can occur. The NRC staff’s evaluations of the analysis methodologies for each FIV mechanism are described below.

3.9.2.4.3.3 Forcing Function Methodologies and Assumed Flow Velocities

The applicant selected TB empirical forcing functions that are most appropriate for the flow and geometries of a given component, such as annular flow for the risers and axial and cross-flow over long beamlike structures (like the CRD shafts and ICIGTs). These forcing function definitions are scaled with geometric and flow variables, such as peak velocity at the center of an annulus flow, and the height of the flow profile. Therefore, flow velocity estimates were needed to compute actual forces and were calculated using the results of the CFD thermal-hydraulic analysis. The CFD calculations were over the full primary coolant region and are based on assumed reactor core and SG power density and loss coefficients. Therefore, spatial variations of the flow through the core and SG were not computed (only bulk velocities are available for those regions). CFD grid refinement studies verified flow velocity convergence throughout the primary coolant flow path. The CFD solution for the highest reactor power and flow conditions was processed to compute average and maximum velocities over several critical cross sections near the components evaluated for FIV. The applicant used the average velocities from its CFD analyses over the cross sections for the TB analyses and VS evaluations of the RVIs. The applicant estimated gap flow velocities for the SG TB, SG FEI and VS analyses based on the geometric blockage of the tubes and the bulk velocity.

All velocities used by the applicant for TB analyses may not be conservative, given the lack of detailed resolution in the CFD models. The applicant's assumptions regarding the location of peak velocity for some components may also be nonconservative. However, other aspects of the applicant's TB forcing function modeling approach, particularly with the parameters chosen in the empirical models (e.g., convective velocities and correlation lengths), are conservative. Given the large margin against TB-induced vibration (due to very low primary coolant flows), it is unlikely that any nonconservative biases in the applicant's assumed peak velocities will lead to significant vibration-induced damage for TB.

Although the primary coolant flow is the main source of FIV in the NPM, turbulent secondary coolant flow will also drive the inner walls of the SG tubes. The secondary coolant enters the SG tubes as preheated water, transitions to boiling on its way to the steam headers, and exits as superheated steam. The applicant used simple turbulent pipe flow empirical models for these forces, but also conducted "separate effects" testing of the wall pressures in the SIET TF-1 test facility (FSAR Section 1.5.1.3, "Steam Generator Thermal-Hydraulic Performance Testing—Electrically Heated Facility"). Strong spectral peaks were observed in the wall pressure data measured in TF-1.

The applicant applied the forces measured in TF-1 to its models of the SG piping in the TF-2 test (FSAR Section 1.5.1.4, "Steam Generator Thermal-Hydraulic Performance Testing—Fluid-Heated Facility"), where tube vibration was measured in the presence of both primary and secondary flow. The calculated TF-2 strains using the TF-1 forces are lower than the TF-2 vibration measurements, showing that the internal forces observed in TF-1 testing do not induce significant vibration (details are provided in NuScale TR-121354). There are also no peaks visible in the TF-2 measurements that are indicative of strong internal flow excitation. Based on this combination of TF-1 and TF-2 measurements and FIV analyses the NRC staff finds that there is reasonable assurance that the secondary coolant flow will not cause adverse FIV effects on the SG tubes.

3.9.2.4.3.4 Structural Mode Shapes and Resonance Frequencies

The ANSYS software suite, which includes structural FE and CFD modeling tools, was used to estimate structural mode shapes and resonance frequencies of the NPM RVIs and SG. FSAR Section 3.9.1.2, states that ANSYS is a pre-verified and configuration-managed FEA program used in the design and analysis of safety related components. The NRC staff finds that NuScale's ANSYS models are acceptable because the applicant has demonstrated that its meshing procedures and spatial resolution, boundary condition assumptions, and fluid loading effects are appropriate and conservative.

External (primary coolant) fluid mass loading was not modeled explicitly; instead, it was assumed to be that of the volume displaced by a given structure, which is a reasonable bounding approximation per FE modeling practices. The secondary coolant mass densities were also added to those of the SG tubes to compute the in-service resonance frequencies, which is also bounding. The applicant modeled all RVIs as an assembly and confirmed the appropriateness of the meshing density with convergence studies. Some structures, like the ICIGTs and control rod drive system (CRDS), were modeled individually using assumed boundary conditions at adjacent structural locations. The NRC staff finds the structural FE modeling reasonable because in general, conservative boundary conditions were assumed for these individual models (leading to lower resonance frequencies, which is conservative when assessing lock-in FIV mechanisms).

3.9.2.4.3.5 Structural Damping

Damping of all RVIs is assumed to be less than or equal to 1 percent, which the staff finds acceptable as it is in accordance with RG 1.20. However, the applicant assumed 1.5 percent damping for the SG tube VS and FEI analysis but did not provide validated test data to substantiate this increased damping. The higher damping is assumed to be caused by friction between the tubing and tube supports, which depends on the tightness of fit, which in turn depends on thermal expansion and operational loads on the tubes and tube supports at normal plant operating conditions. The higher assumed damping led to higher estimated margins against VS and FEI occurring in the SGs. However, NuScale performed testing in the SIET TF-3 facility in Summer and Fall 2024 and showed that VS and FEI do not occur in the TF-3 under tight SG tube to tube support conditions. Representative data from this testing were provided for staff review in October-December 2024 and a final test report summary docketed in January 2025 (ML25027A395 (proprietary) and ML25027A394 (non-proprietary)). Since no VS or FEI were observed in the testing, the higher assumed damping for the initial screening calculations is irrelevant.

3.9.2.4.3.6 Turbulent Buffeting Analyses

The applicant evaluated TB-induced vibration of components with fundamental resonance frequencies below 200 Hz, which the NRC staff finds reasonable since the TB loading above 200 Hz is negligible. Since TB loads are spatially and temporally random, random forced response analysis methods are used with conservative estimates for convection velocity and turbulence integral length scales. The calculated vibration and alternating material stresses are very small due to the low primary coolant flow speeds.

Fatigue and wear due to impacts were estimated for components with nonnegligible TB-induced vibration amplitudes and, in particular, for cases with high relative motion between components and neighboring supports. These include the CRAGT on the CRAGT support, the CRD shaft impact on the alignment cone, the upper ICIGT impact on the second highest CRD shaft supports, and the lower ICIGT impact on the upper core plate. The SG tubes are also assessed for impact wear, but the steady pressure forces on the tubes are expected to maintain nearly constant contact between the SG tubes and tube supports, minimizing the number of impacts that occur over service life.

Peak relative vibration amplitudes were assumed to be 5 times the predicted root mean square amplitudes, which capture a statistically appropriate number of peak occurrences that the NRC staff finds reasonable, based on guidance in Au-Yang's workbook. Previous applicants have extensively referenced Au-Yang's workbook for the FIV analyses, and therefore, the NRC staff finds that the guidance in this reference is acceptable. The number of impacts is based on the average crossing frequency, estimated using well established methods. Worst case contact and estimates for all evaluated components are negligible.

In Section 2.3.3.1 of TR-121353 the applicant discusses the potential FIV effects of the 16 small upper riser slots which provide a flow path for boron redistribution. The slots are small enough to not affect the structural modes of the upper riser significantly. Also, although the slots introduce stress concentrations in the upper riser, the alternating stresses in the upper riser walls induced by TB are so small that the safety margin against the material fatigue endurance limit is not challenged. Finally, the turbulent jet flow through the riser hole may impinge on some SG tubes. However, the slot heights and inclination angles direct any through flow downward to minimize flow-induced loads on the SG tubes. The staff finds that the structural integrity of the riser will not be significantly affected because the holes are small and the safety margin against TB remains very high. Additionally, the NRC staff finds that the jet flow loads will not significantly affect the SG tubes because jet flow loads are low.

The NRC staff finds that the applicant's TB assessments of the NPM RVIs and SG are based on appropriate modeling procedures, assumptions, and inputs, and are reasonable and conservative. No significant TB-induced degradation of RVI or the SGs is expected.

3.9.2.4.3.7 Flutter and Galloping Susceptibility

The applicant examined the shapes and cross sections of any structure subjected to cross-flow and compared them to guidelines for avoiding F/G. These guidelines are well established in the open literature and are acceptable. All NuScale components have significant margin against F/G, and any bias errors in velocity estimation will not challenge the margins; therefore, the NRC staff finds the F/G analyses to be acceptable.

3.9.2.4.3.8 Vortex Shedding and Fluid Elastic Instability Susceptibility

Although there may be some risk of structural wear caused by TB, FIV risks are much higher for stronger mechanisms like VS and FEI. If the frequency of VS aligns with those of structural resonances and if the impedances of those resonances are small, lock-in can occur and cause significant vibration and damage. Structural impedance at resonance is related to the mass-damping parameter in ASME Code, Section III, Appendix N. In addition, if velocities are high enough to induce FEI in arrays of tubes (like the SG tubes), even higher vibrations and damage could occur. All components subjected to cross-flow were screened for susceptibility to VS. The only components that warrant additional evaluation against VS/lock-in are the lower regions of SG tubes. All other components were designed to ensure that the VS frequencies are well below any structural resonance frequencies.

Only the lower SG tubes are subject to VS/lock-in since the primary coolant flows downward, and the lower tubes have no downstream structures to break up the shed vortices. However, all SG tubes may experience FEI at and above critical flow velocities. FEI is therefore evaluated throughout the SGs. The applicant acknowledged that these components need validation testing to ensure margin against these mechanisms, and reported the following margins in TR-121353, in accordance with ASME BPV Code, Appendix N:

- lower SG tube VS/lock-in: **[[ ]]** percent
- SG tube FEI: **[[ ]]** percent (steam region) and **[[ ]]** percent (feedwater region)

The margins are for primary coolant flow rates at 100 percent power where **[[ ]]** margin implies plant power would need to increase **[[ ]]** above 100 percent to induce FEI in the SG. The VS margins are negative, implying VS could occur near 100 percent power. However, NuScale and the NRC staff notes that the VS margins are based on conservative assumptions in ASME BPV Code, Appendix N, Criterion A, which assumes a Strouhal Number (fD/U, where f is frequency, D is tube diameter, and U is flow velocity) of 1.0. Experimental evidence supports a much lower Strouhal Number (0.3 is typical). The actual best estimate margin of safety for VS using a Strouhal Number of 0.3 is **[[ ]]** percent for the lower SG tubes.

The method used by the applicant for FEI assessment is consistent with those in the ASME BPV Code Appendix N. However, the analysis inputs to the method were not conservative. The assumed damping for FEI analysis is 1.5 percent instead of the traditionally accepted 1.0 percent, thereby increasing the mass-damping parameter (increasing margin against FEI). Also, the so-called Connors constants used by NuScale to assess susceptibility to FEI (C=1.9, a=0.05) are not typical and deviate significantly from commonly accepted values (C=2.4, a=0.50 recommended in Section N-1331.3, “Suggested Inputs” of Appendix N of ASME BPV Code). The staff estimates that if 1 percent damping and the ASME recommended Connors constants were used the NPM SG would have no margin against FEI at 100 percent power.

Therefore, some form of testing was needed to confirm that FEI will not occur in the NPM. Testing was performed in the SIET TF-3 facility in Summer and Fall 2024 at flow conditions spanning very low to very high power (well above 100 percent equivalent NPM power). The staff examined preliminary test data both on-site at SIET in October 2024 and at an in-person audit at NuScale October 22-24, 2024. The staff concluded that the test facility reasonably represented the flow-induced vibration behavior of an NPM SG. In particular the tube-to-tube support connections were tight during flow testing and the corresponding damping of tube modes was small (on the order of 1 percent which is consistent with RG 1.20 Revision 4 guidance). Therefore, any differences between the TF-3 tube support design and that of the NPM-20 have no impact on the TF-3 test results (both are expected to be tight fitting). A preliminary data analysis report was provided in December 2024 and confirms that neither VS nor FEI occurred in the testing. A summary of the final report was submitted in January 2025 and further confirms that VS and FEI are not expected to occur in the NPM.

3.9.2.4.3.9 Acoustic Resonance Susceptibility

AR issues in nuclear power plants are usually associated with flow instabilities that form over side openings in the pipe flow. The fundamental acoustic modes in valve standpipes are the most commonly excited resonances. ARs have occurred in existing nuclear power plants and have led to extensive damage to valves and RVIs, particularly in boiling-water reactors. The flow instabilities occur when a half or full wavelength of the vortices shed from the leading edge of a side branch and coincide with the diameter of the opening. The first order (half-wavelength) instability is strongest and is most likely to lock in to any acoustic modes within the side branch. However, strong second order (full-wavelength) instabilities can induce damage to the valve components and other RVIs.

The applicant has evaluated the following piping and valve components for susceptibility to the first and second order AR, including components in the CNTS. Analysis margins against the first order AR are shown in parentheses for each component.

- RCS injection line to ECCS reset lines (**[[ ]]** percent)
- CNTS RPV high point degasification drain valve branch (**[[ ]]** percent)
- CNTS FW drain valve branches (**[[ ]]** percent)
- SG system pressure-relief valve branches (**[[ ]]** percent)
- DHRS condensate line to SGS feedwater line (**[[ ]]** percent)

Due to adherence to the best design practices for AR avoidance, including rounding of the cavity upstream edges where possible, no NuScale piping or valve components are expected to experience first-order instability AR at full plant power conditions (all have more than 100 percent margin). However, these components might experience second-order instability AR at less than the full plant power level:

- DHRS condensate line to SGS feedwater line with **[[ ]]** percent margin
- CNTS RPV high point degasification drain valve branch with **[[ ]]** percent margin

Two locations evaluated in the DCA with initially low AR margins were the CNTS MS branch connection to the DHRS steam piping and the DHRS MS drain. Flow disruptors will be installed at the leading edges of the side branch openings at these locations to mitigate possible AR. As noted in FSAR Table 14.2-102, “NuScale Power Module Vibration Test # 102,” these locations will be monitored during the initial startup testing (to ensure that the vibrations at these plant power levels are not excessive). The NRC staff concluded that the applicant’s AR analysis methods and calculations are based on validated methods, and there is margin against both first- and second-order AR. Components treated with leading edge spoilers will be tested during the initial startup. Therefore, the NRC staff finds that there is reasonable assurance that significant AR-induced vibration will not occur, and that if AR occurs it will be detected so that changes could be implemented to preclude damage.

In FSAR Section 3.9.5.1, the applicant evaluated the through holes in the upper riser for susceptibility to AR effects. The upward flow in the upper riser and the downward flow in the SG annulus pass over the holes, which could induce shear flow instabilities and could potentially generate appreciable pressure pulsations within the primary coolant. However, the pressure difference between the upper riser and SG annulus drives a modest amount of flow through the hole, which eliminates the possibility of flow instabilities. Nevertheless, the applicant also compared the possible range of flow instability frequencies to those of acoustic modes within the upper riser coolant. The frequencies are far apart, eliminating the possibility of a flow instability driving an AR. The NRC staff conducted an audit during the NuScale DCA review of the applicant's evaluations (see audit report ML20160A247) and found that the riser holes would not cause AR because there is flow through the holes and the flow instability frequencies and the upper riser acoustic frequencies are well separated.

3.9.2.4.3.10 Leakage Flow Instability Susceptibility

Fluid-dynamic forces induced by leakage flow in the gaps between a structure and an external passage can couple with translational and rotational modes of the structure, sometimes to the point where self-excitation or lock-in occurs. Self-excited vibration amplitudes can be very high and cause contact between the structure and passage. Over time, repeated contact can cause wear and/or material fatigue damage. Damaging LFI has been observed previously in commercial nuclear reactors (M.P. Paidoussis, "Real-life Experiences with Flow-Induced Vibration," *Journal of Fluids and Structures,* Volume 22, pages 741–755, 2006), and design guidance has been developed for its avoidance (T.M. Mulcahy, "A Review of Leakage Flow Induced Vibrations of Reactor Components," Argonne National Laboratory Report ANL-83-43, May 1983).

The applicant has designed its components using best practices for LFI avoidance. In particular, there are no diverging passages between components, nor are sudden structural expansions located at the entry to a passage. Also, most components with leakage flow paths have very low-pressure differentials to ensure that leakage flow rates are small. As with the other instability mechanisms investigated by the applicant, a critical flow velocity is estimated and compared to the localized velocity at full plant power conditions. Margin is based on the ratio of the critical to localized velocity.

The applicant evaluated the following RVI components for LFI using the methodology defined in the TR-121353 references (Inada, 1990, 1990, 2015):

- CRD shafts adjacent to all through holes in surrounding support structures
- CRD shaft sleeve
- ICIGT adjacent to all through holes in surrounding support structures

The NRC staff evaluated the LFI evaluation procedures in the applicant's cited references and performed sample confirmatory calculations. The NRC staff finds that the procedures are reasonable and validated against test data, and the NRC staff's confirmatory calculations are consistent with the applicant's calculations (provided during a 2019 audit for the NuScale DCA review; ML19340A015).

The LFI evaluation methodology requires knowledge of the pressure difference across a passage and the loss coefficients for flow into and out of a passage. The pressure differences were estimated from the applicant's CFD analyses. The loss coefficients were estimated using standard thermal-hydraulic methods. Structural damping was assumed to be 1 percent, which is consistent with RG 1.20. The applicant also conservatively assumed a slightly diverging **[[ ]]** annular gap area increase in all passageways to account for manufacturing tolerance uncertainties (even though this is unlikely to occur). Critical velocity was estimated as the point where total effective damping becomes negative (where LFI effects cancel the 1 percent structural damping). All components have more than 100 percent margin against LFI. Since there is significant margin, there is no need for testing prior to the initial startup.

Despite the high estimated margins against LFI for forward flow, pressure sensors will monitor the acoustic field of the RPV during the initial startup testing of the first NPM reactor to ensure that no unexpectedly high vibrations occur due to LFI or any other FIV mechanism (per FSAR Table 14.2-102, "Test # 102 NuScale Power Module Vibration" and Section 6.0, "Initial Startup Measurement Testing" of TR-121354). The NRC staff concluded that the applicant's LFI analysis methods and calculations are based on validated methods, there is significant estimated margin against LFI, and those components are part of the post-initial startup inspection plan. Therefore, the NRC staff finds that there is reasonable assurance of no significant LFI-induced vibration and structural damage for the life of an NPM.

3.9.2.4.3.11 Density Wave Oscillation Instability

A DWO instability in the secondary coolant within the SG tubes would lead to slow oscillations of the boundaries between the inlet subcooled liquid and the two-phase region and outlet steam. For mild small amplitude DWO instability, some of the subcooled liquid in the lower SG tubes will flow backwards through the IFRs with no significant thermal or structural loading. In severe high amplitude DWO conditions however, the boiling boundary, normally well above the SG tube inlets, and even steam could flow backwards through the IFRs, leading to potentially strong and sudden transient loads. NuScale has specified the primary and secondary steady state parameters to minimize the likelihood of DWO onset, as well as the possibility of severe DWO. During normal operation at all power levels Figure 4-9, "SG Collapsed Liquid Level," of EC-110662 Revision 2 (see response to Audit Question A-5.4.1.3-3, Figure 18) shows that the boiling water boundary should remain far from the SG tube inlets (at least 30 percent of tube length) and **[[**

]], minimizing the chance of any two-phase flow inducing cavitation or CIWH loads on the tube inlets or in the FW plenum. The SG tube inlets are also fitted with IFRs which add significant resistance to any oscillatory subcooled liquid flow, further stabilizing the SG system.

Although the operating conditions and IFRs will limit the conditions under which DWO instabilities can occur (those conditions are during specific transient events as outlined in ER-101144, Revision 3 included in the response to SDAA Audit Question A-3.9.2-28 (ML24346A152 (proprietary), ML24346A151 (non-proprietary)), primarily when the DHRS is engaged), NuScale has stated that the RVI and SGs can withstand up to 2840 days of operation at DWO over the life of a plant. The plant operational time in conditions where DWO might occur will be tracked by monitoring the "approach temperature" (defined in FSAR Section 5.4.1.3) – the difference between the primary and secondary coolant temperatures at the SG outlet. Thermal-hydraulic simulations of the NPM-20 SG at a wide range of operating conditions show that **[[ ]]** when the approach temperature is small or nearly 0 (as shown in the example in Figure 3-4, "SG Tube Fluid Normal and DWO Onset Temperature Profiles" of EC-174500, Revision 1 (see response to Audit Question A-5.4.1.3-3, Figure 22). A small approach temperatures imply **[[**

]]. SIET TF-1 data, however, showed that **[[**

]], and are included in NuScale's DWO monitoring approach described in FSAR 5.4.1.3. These conditions, where the boiling boundary **[[** **]]**, should not lead to any mechanical loads on the tube inlets, IFRs, or tube sheet.

Any plant operations where the approach temperature is lower than acceptable limits (shown as a function of plant power in FSAR Figure 5.4-16, "Approach Temperature for NPM-20") is counted as potential time at DWO regardless of whether the actual DWO onset has been reached. **[[**

]].

The ability of the approach temperature limit to provide reasonable assurance of protection against onset of DWO, along with the appropriateness of NuScale thermal-hydraulic modeling of the locations of **[[** **]]**, is evaluated in Section 5.4.1.3 of this SER. As an added fail-safe, NuScale's calculations show that the NPM-20 is likely to trip if the primary coolant temperature exceeds 555 degrees Fahrenheit (F) (this implies **[[** **]]**). The trip points are well before expected DWO onset for several initiation conditions at and above 25 percent NPM power, implying the reactor will usually trip before DWO onset can occur. The staff finds that the IFRs and the conservative approach temperature limit, the high likelihood that the **[[** **]]**, and limiting the amount of time operating below the approach temperature limit will ensure minimal DWO-induced damage of the SG tubes and IFRs.

Given the unlikelihood of incoherent DWO, the possibility of coherent DWO developing is extremely low. NuScale addressed this possibility in the response to SDAA Audit Question DWO-SC-22 (ML25013A207 (proprietary) and ML25013A206 (nonproprietary)). Should coherent DWO occur it would be observable in system monitoring and considered a system-level instability and very likely mitigated by existing control systems. Operating procedures which describe how to monitor and mitigate system level instabilities like coherent DWO will be developed per FSAR Section 5.4.1.3 and COL item 13.5-3. Therefore, the staff finds that NPM-20 operating with coherent DWO conditions is highly unlikely.

[[**]]** several relevant cases in the open literature discussed above demonstrate clearly that the period of a secondary coolant DWO instability cycle is much longer (usually more than 10 seconds) than the periods of structural resonance frequencies in the SG system and tubes. Therefore, the staff finds that DWO instabilities, should they occur, will not couple strongly with any system resonances and any mechanical loading will be benign.

Although operating at severe DWO is unlikely and operating at low approach temperature conditions will be limited, NuScale considered and assessed the following loading mechanisms associated with severe DWO in their responses to various audit questions:

- Sudden (CIWH) like events impinging on the SG tubes just downstream of their inlets and on the IFRs in the FW plenum

- cavitation loads on the IFRs and IFR mounting systems and SG tube interior walls

- tube wear from slow sliding between the tubes and tube supports induced by thermal gradients

- Fatigue wear on tubes, tubesheet (at the tube inlets), and tube-to-tubesheet welds

In the response to SDAA Audit Question A-3.9.2-34 (ML24346A158 – Proprietary, ML24346A157 – Non-proprietary) NuScale compared thermal-hydraulic conditions in the NPM-20 SG at severe DWO conditions to those which induce water-hammer like loads observed in previous BWRs and PWRs and cited in NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," NUREG/CR-5220, "Diagnosis of Condensation-Induced Waterhammer," and NUREG/CR-6519, "Screening Reactor Steam/Water Piping Systems for Water Hammer." The geometry of the SG tubes (inclined at helical angles) precludes typical CIWH loads. In the unlikely event that large "slugs" of liquid form in the two-phase flow region and accelerate backwards toward the tube inlets, NuScale evaluated bounding impulsive loads based on a peer-reviewed article (Riverin, "Fluctuating forces caused by internal two-phase flow on bends and tees," Journal of Sound and Vibration, 298 (2006) 1088-1098). NuScale determined that the bounding loads are negligible and if they occurred would not lead to tube or other component damage. CIWH in the feedwater plenum was not evaluated given the conservative approach temperature operating limit (as discussed above) Therefore, the staff concurs with NuScale that there is very low probability of two-phase flow through the IFRs and into the FW plenum.

NuScale evaluated the potential for cavitation-induced surface wear in the SG tube inlet region in Section 3.2.9, "Steam Generator Tube Inlet Flow Restrictor Density Wave Oscillation Cavitation Flow Assessment" of the CVAP analysis report (TR-121353-P) and in the response to SDAA Audit Question A-3.9.2-26F (ML24346A150 – Proprietary, ML24346A149 – Non-proprietary) and found it minimal. NuScale also evaluated the possibility of cavitation in the FW plenum at and around the IFR mounting hardware in the response to SDAA Audit Question A-3.9.2-28 (ML24346A152 - Proprietary, ML24346A151 – Non-proprietary). Significant damage to the IFR mounting hardware is highly unlikely. Should any mounting hardware fail, other preloading mechanisms would prevent the IFR from dislodging.

Thermally-induced damage including tube wear caused by sliding against supports and stresses in the tubes, tubesheet, and tube-to-tubesheet welds were evaluated in the response to SDAA Audit Question A-3.9.2-26 (ML24346A148 – Proprietary, ML24346A147 – Non-proprietary). A bounding thermal transient was assumed based on the lower and upper temperature limits of the secondary coolant (to maximize thermally induced deformations) and the longest possible DWO time (which maximizes sliding distances). In a sensitivity study, the maximum number of cycles was assumed based on the 2840 days of allowable operation at low approach temperature conditions and the conservative shortest possible DWO cycle time. Even with these conservatisms wear was less than the allowable limit. In the unlikely event significant wear occurs, it would be observable within NuScale's inspection interval (see below). NuScale shows that thermally-induced stresses are very small with negligible fatigue life impact **[[
]]**.

NRC Office of Research performed confirmatory studies of the NPM SG under various conditions conducive to possible DWO (ML25007A231). The studies used a thermal-hydraulic software – TRACE (https://www.nrc.gov/about-nrc/regulatory/research/safetycodes.html#th) – and showed that severe DWO is highly unlikely. As was shown in NuScale's NRELAP5 simulations the boiling boundaries will remain far from the tube inlets during any DWO event. This implies that CIWH and cavitation near the tube inlet is highly unlikely.

Although severe DWO is highly unlikely to occur, and all of the loading mechanisms above appear to be benign, to ensure any unexpected wear or erosion is detected, FSAR, Section 5.4.1.6.1 and the SG inspection program (Section 5.5.4, "Steam Generator (SG) Program" of the US460 GTS, Volume 1) ensures, in part, that all SG tubes will be visually inspected after the first refueling outage and on a staggered basis every six years (72 effective full power months) afterwards. This inspection will involve removing the IFRs and the IFR mounting hardware, which will therefore also be inspected. Any tubes violating the steam generator tube plugging criteria in the Technical Specifications will be plugged. Section 5.5.4 of this SER evaluates NuScale's planned inspection program along with plugging criteria.

The staff has evaluated:

- The approach temperature monitoring methodology (in this section and in Section 5.4.1.3 of the SER)

- The current primary and secondary steady state conditions in and around the SG, including the estimated heights of the boiling water boundary (with appropriateness of calculation methodology confirmed by the staff in SER, Section 5.4.1.3 under item G2.1, "The evaluation model contains the appropriate modeling capabilities") and steam transition boundary

- The estimated amount of time approach temperature conditions which preclude DWO could be violated

- The possibility of coherent DWO occurring in the SG as well as being induced in the secondary coolant system

- The bounding loads that could be induced during the unlikely event of severe DWO operation including thermal transients, cavitation, and CIWH

- The SG Technical Specification which requires full inspection of both SGs after the first refueling outage and every 72 effective full power months of plant operation afterwards; along with plugging non-compliant tubes (see Section 5.4.1.3 of this SER)

The staff finds the likelihood of strong incoherent (or coherent) DWO occurring to be small throughout the life of the plant. In the highly unlikely event the NPM-20 operates in a sustained incoherent DWO state, the bounding loads are not expected to induce thermal or structural fatigue or tube failures within a 72 effective full power month period. Any unexpectedly high and excessive damage should be discovered during inspections and mitigated by plugging tubes. Should coherent DWO occur, NuScale would consider it a system level instability and adjust plant operating parameters to mitigate it.

3.9.2.4.3.12 Benchmarking Testing

The applicant performed limited testing to benchmark its FIV analysis methodologies and relied more heavily on screening and analysis results to identify RVI, piping, and SG components that are at risk of damage resulting from FIV and to identify the analysis areas that require subsequent validation testing. Benchmark testing was performed for the SG using:

- SIET TF-1 secondary flow testing
- SIET TF-2 modal testing
- SIET TF-2 primary and secondary flow testing
- SIET TF-3 ”build-out” modal testing
- SIET TF-3 flow testing

TR-121354 describes the results of the benchmark tests with the exception of the flow testing which occurred in Summer and Fall 2024.

Some resonant peaks exist in TF-2 SG tube vibration spectra during flow testing, along with unexpected strong forces induced by two-phase secondary flow within the tubes (TF1 testing). However, the mild variation of TF-2 vibration peaks with increasing flow is not indicative of VS or FEI behavior (where vibration can increase substantially due to minor flow changes). Also, the unexpected TF-1 forces, when applied to models of the SG tubes, do not induce significant vibration. Finally, simulations of the TF-2 vibrations using the TB tools applied to the full-scale plant FIV analyses were shown to be conservative when compared to measured data.

NuScale’s NPM SG VS and FEI assessments assumed tube damping ratios of 1.5 percent which is higher than the 1 percent allowable in RG 1.20 without confirmatory testing. Modal testing of the “build-out” in-air configuration of TF-3 shown in Section 3, “Benchmark Testing” of TR-121354 shows damping levels lower than the assumed 1.5 percent (in fact nearly identical to the usually accepted 1 percent). However, successfully demonstrating that VS or FEI cannot occur in TF-3 with on the order of 1 percent damping resolved this issue as shown in the final TF-3 test report summary issued in January 2025.

Finally, no benchmark testing was performed to assess the possibility of AR in the steam system since margins are expected to be much higher than 100 percent. Side branches with flow disruptors installed will be instrumented during the initial startup testing of the prototype NPM. These planned tests are evaluated in Section 3.9.2.4.4 below.

3.9.2.4.4 Flow-Induced Vibration Validation Testing and Inspection of Reactor Vessel Internals and Steam Generators

The planned measurement program details for validation testing of the NPM RVI and SG system are provided in TR-121354, Revision 1. A second report (to be submitted after the SDAA certification) will include the post-measurement evaluations and will be submitted after the completion of the validation testing and after the initial startup testing.

Validation testing was performed on the following:

- Prototypic SG tubes without secondary coolant with near-prototypic supports (SIET TF-3 for modes, damping, VS, and FEI) as described in FSAR Section 14.2, Test #65

According to Section 4.0, “Vibration Measurement Program,” of TR-121353, the initial startup testing (FSAR Section 14.2, Test #102) will be performed on the following:

- any RVI with less than 100 percent margin against a significant FIV mechanism
- selected sections of the CNTS piping for AR, including novel design changes such as flow disrupters

General acceptance criteria are defined for the prototypic SG validation testing in TR-121354 and in Table 14.2-65, “Test # 65 Steam Generator Flow-Induced Vibration” of FSAR Chapter 14.2 and are:

- The SG tube testing shows that FEI and VS do not occur under primary side flow rates consistent with any operating condition, considering all applicable uncertainties and biases of this separate effects test.
- The SG tube testing shows that for primary side flow rates consistent with 100 percent power operation, the SG tube vibration responses are less than those predicted with the TB analysis methodology.

General acceptance criteria are defined for the NPM prototype vibration testing in TR-121354 and in Table 14.2-102, “Test # 102 NuScale Power Module Vibration” of FSAR Chapter 14.2 and are:

- Measured vibration amplitudes in the CNTS steam piping branches confirm there is no AR concern.
- Measured vibration responses in the NPM confirm there are no resonant peaks that could indicate a strongly-coupled flow induced vibration mechanism.

Development of the detailed acceptance criteria for NPM initial startup testing is deferred to the COL applicant. COL Item 3.9-4, states that a COL applicant will provide the applicable test procedures before the start of testing and will submit the test and inspection results from the CVAP for the NPM, in accordance with RG 1.20. To ensure that the acceptance criteria are appropriate and corrective actions will be taken if the criteria are violated, the applicant committed (in FSAR Section 14.1 and 14.2), to meet Criterion XI, “Test Control,” of 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.” The NRC staff finds that COL Item 3.9-4 provides reasonable assurance that the COL applicant will establish appropriate test acceptance criteria and the 10 CFR Part 50, Appendix B requirements ensure that the test acceptance criteria are met.

The NRC staff’s detailed evaluations of each test are provided in the sections below.

3.9.2.4.4.1 Final Design Testing of the Steam Generator Tube Inlet Flow Restrictors

There are no plans to perform final testing of the steam generator tube inlet flow restrictors since the simple design does not require additional verification by testing due to very high margins against damaging FIV.

3.9.2.4.4.2 Flow-Induced Vibration Testing of Steam Generators in Test Facility 3

The susceptibility of the SG to VS and FEI was evaluated prior to the initial plant startup in the SIET TF-3 test facility. Evaluating a reasonably prototypic SG in a separate facility allowed more rigorous measurements with extended instrumentation and over much higher flow rate conditions than those possible in the actual plant. The expanded instrumentation and higher flow rates (over two times that planned for an operating NPM) allowed NuScale to more conclusively establish the ranges of operating conditions under which VS and FEI will and will not occur.

The applicant specified in Section 5.1, "TF-3 Validation Test" of TR-121354 requirements for the test facility, instrumentation, data acquisition, and operation, along with a list of planned tests. The NRC staff also audited the TF-3 preliminary test plans (NuScale internal document TSD-T050-54312, Revision 7, in response to Audit Question A-3.9.2-24 (ML24346A146)).

TF-3 flow testing was conducted in Summer and Fall 2024 and observed by the staff at SIET in October 2024. Preliminary staff assessments of measured tube vibrations showed no evidence of loose connections. Damping loss factors estimated from tube vibrations under flow testing were in the range of 1 percent. The TF-3 SG assembly therefore appears to be reasonably representative of that of an NPM with tight tube to tube support interfaces.

To establish allowable ranges of critical flow velocities at which VS or FEI may occur in TF-3, the applicant performed an uncertainty analysis. The uncertainty analysis is based on procedures described in the ASME Standard for Verification and Validation in Computational Fluid Mechanics and Heat Transfer and implemented as discussed in TR-121354, Section 4. The procedures are reasonable and include uncertainty in modeling and analysis (used to estimate the actual in-plant margins against VS and FEI), measurement methods, and differences between the TF-3 and the NPM with one exception discussed below. The applicant computed these total uncertainties for both VS and FEI as shown in Section 5 of TR-121354. Allowable ranges of flow velocities are specified for both mechanisms in TF-3 in Tables 5-18, "Vortex Shedding Test Range" and 5-19, "Fluid Elastic Instability Flow Tests." The staff estimates that the lower allowable limits for critical FEI and VS flow velocities in TF-3 correspond to **[[ ]]** percent and **[[ ]]** percent margins compared to equivalent 100 percent flow conditions in the NPM. The applicant acknowledged that testing uncertainties may decrease in the future. Such decreases will effectively increase the margins against FEI and/or VS and, therefore, the NRC staff finds this to be acceptable.

TB-induced vibration of SG tubes was also measured in TF-3. The predicted vibration levels and corresponding alternating stresses are very low. Nevertheless, the applicant provided pretest predictions of maximum allowable TB-induced vibrations. Based on the benchmarking studies in TF-1 and TF-2, the NRC staff finds that the predictions are expected to be conservative and are reasonable. The NRC staff also finds that the applicant's uncertainty analysis captures the important analysis and measurement variables and is appropriate.

NuScale submitted their final TF-3 test report summary in January 2025. The report shows:

- The tube damping at flow rates up to about 200 percent full power is small and indicates tight tube-to-tube support connections, showing the test facility is appropriate for screening for possible FEI and VS (loose connections would have induced non-prototypic high damping which would have prevented VS or FEI from occurring).

- There is no evidence of VS or FEI in any of the flow tests. Measured tube vibrations show resonance peaks at frequencies consistent with those in FE simulations. The peak vibrations increase with increasing flow at a rate typical of TB for speeds up to 250 percent full power.

- At very high flow speeds the tube vibration is strong enough to induce acoustic pressures in the water that propagate to the external pressure sensors mounted on the outer walls. The external pressure sensors clearly show the tube vibrations, indicating they are suitable for monitoring high internal structural vibrations.

Based on the TF-3 test data the staff concludes that VS and FEI is not likely to occur in the NPM.

3.9.2.4.4.3 Initial Startup Testing of NPM

Since there are no NPM components that are expected to experience excessive FIV, planned initial startup testing of the prototype NPM for FIV mechanisms of RVIs is limited to:

- performing a flow test of the CNTS MS piping to confirm the lack of significant AR, and
- identifying and localizing any unexpectedly strong FIV effects in RVIs and the SG.

A short summary of the test method is provided in FSAR Table 14.2-102:

- Perform load ramp up to 100 percent power, then operate the NPM for a sufficient duration at 100 percent power to ensure one million vibration cycles for the component with the lowest structural natural frequency.

- Monitor the vibration of the CNTS steam piping branches, including the DHRS steam lines and MS drain valve branches. Also monitor the signals of the dynamic pressure sensors. If an unacceptable vibration response develops at any time during initial startup testing, the test conditions must be adjusted to stop the vibration and the reason for the vibration anomaly are investigated before continuing with the testing.

More details are available in TR-121354 and additional details will be provided by the COL applicant consistent with COL item 3.9-4.

Three instrumentation suite options have been proposed in Section 6.4 of TR-121354 to monitor the vibrations of RVI and the SG. All options use 14 dynamic pressure sensors with three candidates listed in Table 6-9, "Candidate Dynamic Pressure Sensors" which can withstand the temperatures and pressures within the NPM. No accelerometers or strain gages are recommended. The three options have different numbers of internally and externally mounted pressure transducers:

- Option 1: 2 internally mounted and 12 in RPV shell
- Option 2: 10 internally mounted and 4 in RPV shell
- Option 3: 14 internally mounted

Each option specifies sensor locations intended to monitor significant FIV at and near the:

- Upper riser and top of SG (4 internally or externally mounted)
- Middle of SG tube bundle (4 internally or externally mounted)
- Bottom of the SG and downcomer (4 internally or externally mounted)
- Upper riser (2 internally mounted on upper riser hanger plate)

All sensors are intended to be mounted such that the sensing element is nearly flush with the inner diameter of the mounting surfaces

NuScale has estimated the acoustic pressures which would be measured by the different sensors due to SG tube motion and CRDS motion, including the potential for metal-to-metal contact. NuScale believes significant FIV should be measurable by all pressure sensors.

The adequacy of using only pressure transducers for monitoring FIV (without any internally mounted accelerometers or strain gages) was benchmarked using the SIET TF-3 test facility. TF-3 is instrumented with both pressure transducers as well as vibration sensors on the tubes. At high flow speeds TB of the tubes was audible in the external pressure sensors at tube resonance frequencies as shown in the test report summary. Therefore, the staff concurs that NuScale's proposed external pressure sensor suites should be adequate to detect any unexpectedly high RVI or SG vibrations. TR-121354 also provides specifications for testing CNTS MS line branch connections (DHRS steam piping tees, MS drain valve branches) to assess any significant AR effects. Flow disrupters will be installed at both locations to minimize the chances of AR. Because any strong AR will lead to high vibrations and internal pressures, both are monitored. Several accelerometers will be mounted to the piping and branch connections to monitor vibration. Pressure taps (which will penetrate the piping to directly measure pressures) or circumferential arrays of externally mounted strain gauges (which indirectly measure pressure through hoop strain) will be installed. The NRC staff finds that the measurement procedure is acceptable because the pressure taps measure the pressures directly, and strain gauge arrays have been used successfully in many boiling water reactor MS measurements during extended power uprates. Both MS lines will be instrumented. This combination of instrumentation is sufficient to determine whether significant ARs are present.

Provisions for NPM initial startup vibration testing procedures and data acquisition have not yet been provided and will be submitted prior to the initial startup testing based on actual instrumentation and acquisition systems to be used. The NRC staff finds that the frequency ranges specified are reasonable, consistent with measurement programs used in previous plants, and should bound any significant resonant and/or FIV peak frequency responses, including metal to metal impacts. The testing duration was determined based on the lowest structural natural frequency from the analyses, and a goal of 1 million cycles of vibration is to be achieved, which is acceptable and consistent with common practice. AR testing will include varying flow rates to ensure significant AR does not occur across the full range of operating conditions. AR testing will also comply with the ASME Standard for Operation and Maintenance (OM) of Nuclear Power Plants Part 3: "Vibration testing of piping systems."

3.9.2.4.4.4 Inspections

In accordance with TR-121354, Section 7, "Inspection Program," NPM components that were evaluated for FIV will be inspected after the initial startup testing, following the guidelines and requirements provided in ASME BPV Code, Section III, paragraph NG5111, "General Requirements," and paragraph NB-5111, "Methods," and using the methods defined in ASME BPV Code, Section V, "Nondestructive Examination." VT-1 and VT-3 will be used to perform the visual inspections, as defined by ASME BPV Code, Section XI, Subarticle IWB2500, “Examination and Pressure Test Requirements,” Table IWB-2500-1 (B-N1, B-N2, BN3), “Examination Categories BN1, Interior of Reactor Vessel; B-N2, Welded Core Support Structures and Interior Attachments to Reactor Vessels; B-N3, Removable Core Support Structures.” The inspected areas include major load-bearing elements of the RVIs, restraints inside the RPV, locking and bolting components whose failure could affect the RVI integrity, contact surfaces, critical locations identified by the analysis program, and the RPV interior for loose parts. Visual examinations will be performed to assess the evidence of (1) cracks, defects, or abnormal distortion on critical surfaces, (2) cracks on welds, (3) wear, distress, or abnormal corrosion on interface surfaces, and (4) looseness of fittings. The applicant also plans periodic ISIs of the RVIs per FSAR Section 5.2.4.1. The SG program of the US460 GTS, Volume 1, specifies approximately six-year (72 effective full power months) inspection intervals after the first refueling outage. The NRC staff finds that the inspection methods and areas are consistent with those in previous applications and with the guidance in RG 1.20.

3.9.2.4.5 Dynamic System Analysis of the Reactor Vessel Internals and Steam Generators under Service Level D Conditions

This section contains three subsections. The first subsection contains the NRC staff’s evaluation of the calculated loads on the NPM from postulated seismic events. The second subsection contains the NRC staff’s evaluation of the calculated loads on the NPM induced by postulated short-term transient events. The third subsection is the RVI components stress analysis under the Service Level D faulted conditions.

3.9.2.4.5.1 Seismic Analysis of NuScale Power Module

TR-121515 documents the NPM seismic analysis. The technical report contains analysis methodology, input motion, structural modeling of the major NPM components (i.e., the containment, reactor vessel, upper RVI, lower RVI, CRDM, and TSS) and analysis results, including displacements, ISRS, forces, and moments at component interfaces. The major NPM components were modeled by ANSYS FE meshes. The calculated component interface displacements, forces, and moments were used as inputs for the component-level stress analyses of RVI and the SG tubes. This subsection evaluates the analysis methodology, structural modeling of the NPM (including the RVI and SG), and analysis results as they pertain to ASME Service Level D assessments of RVI and SG stresses. Assessments of the seismic modeling of structures other than the NPM are in Section 3.7 of this SER.

3.9.2.4.5.1.1 Analysis Methodology

The NPM seismic analysis methodology consists of the following steps:

1. Analyze a model of the entire DB which includes the RXB, the RWB, and the engineered backfill using postulated seismic ground motion spectra and several soil profiles (these analyses are reviewed in Section 3.7 of this SER). The analysis uses procedures in TR-0118-58005-A, Revision 2, “Improvements in frequency domain soil-structure-fluid interaction analysis” (ML20353A439) and assumes six NPMs are installed. Linearized simplified NPM (NPM-SE) models are used in the RXB model and coupled with both the structural motion of the building at various support locations and with the water in the pool. NuScale compared analyses of the DB using the simplified and detailed NPM (NPM-DM) models in Appendix A of TR-121515-P, Revision 1. The bounding ISRS at several locations in the RXB and RWB are nearly identical for

both NPM models. Therefore, the staff concurs that the simplified NPM model is suitable for DB seismic analyses.

2. Analyze the detailed NPM (NPM-DM) model using bounding time histories of the accelerations computed from the DB model as boundary conditions. **[[ ]]** of time history are used which includes at least **[[ ]]** percent of the earthquake energy. The individual NPM analyses are nonlinear, allowing for contact and nonlinear material response. In-structure displacement time histories, bounding response spectra, and bounding load amplitudes are computed for subsequent ASME stress analyses of RVI and the SG tubes. The same general NPM model is used for both analyses. All analyses were performed with ANSYS. The staff finds the analysis methodology is consistent with acceptable practice and is acceptable. NPM component seismic analysis uses **[[ ]]**. RG 1.61, Table 6, “Damping Values for Mechanical and Electrical Components,” recommends 3 percent SSE damping for pressure vessels and major pressure boundary components. However, the NRC staff finds that using **[[ ]]** damping in NPM subsystems and system analysis is reasonable and acceptable. The integrated NPM with many connections and internal structures is unlike traditional shell type pressure vessels, and use of damping higher than the 3 percent damping value listed in RG 1.61, Table 6, is reasonable based on the additional energy dissipation provided by the connections and internal structures. Also, the Rayleigh damping method actually induces less than the target **[[ ]]** damping at frequencies between the “grounding frequencies.” In the NuScale analyses damping is less than **[[ ]]** the most important frequency ranges for different soil types. The staff finds the NuScale assumed damping reasonable.

3.9.2.4.5.1.2 NuScale Power Module Modeling

The NPM model, used in both the DB model in linearized form, and for detailed single NPM nonlinear analyses, includes the CNV, piping inside containment, TSS, RPV, LRVI and URVI, CRDM, and CRDM support structures, the fuel assembly, and pool bay and walls. Pool water is modeled with ANSYS fluid elements and coupled to neighboring structural surfaces. All important load transmission paths are included in the model, including the main load transmission paths to the RVI through the upper hanger plate and the upper and lower core plates (UCP and LCP). Transmission paths into the SG are also included through the FW inlet and steam outlet and the radially oriented set screws which push through the upper riser walls into the backing strips of the SG supports. Coupling through the fluid annulus between the RVI and RPV walls is modeled.

The RVI is subdivided into upper (URVI) and lower (LRVI) sections. The URVI includes the upper riser, CRAGT and CRAGT support plates, CRDS and CRDS support plates, and the upper hanger plate. The LRVI includes the lower riser, core barrel, reflector blocks, fuel assemblies, LCP, and UCP. The water inside the RVI is modeled as simple added mass. The URVI rests on the LRVI along an annular interface. Pins are used to ensure the interface is secure. A bellows section in the URVI just above the interface allows for some relative movement between the two sections. The bellows design and stiffnesses are not yet finalized so NuScale performed sensitivity analyses of the dynamic behavior of the RVI assembly with ranges of expected bellows stiffnesses in Appendix B of EC-170084, “ASME Service Level D Finite Element Evaluation of the RVI” (referenced in response to RAI 10111, Question 3.9.2-1, Revision 1 (ML24353A030 (proprietary) and ML24353A029 (nonproprietary))). The results show little difference across the expected stiffness ranges and the staff concurs with NuScale’s assessment that the RVI model is insensitive to the expected range of bellows design stiffnesses.

The staff finds that the modeling approaches, models, and interconnections between models are consistent with common practices and ANSYS is an accepted FE analysis code for nuclear power plant linear and nonlinear dynamic response analysis.

3.9.2.4.5.1.3 NPM Reactor Vessel Internals and Steam Generator Seismic Load Analysis

TR-121515 provides examples of ISRS at the following locations to support subsequent structural integrity analyses of RVI and SG components:

- Core plate
- LRVI
- URVI
- CNV
- RPV and CRDM supports

Bounding forces and moments over all soil types and NPM locations were extracted from the time histories to perform engineering calculations of stresses in fasteners and interfaces between components as well as simple structures. The staff finds these forces and moments to be conservative since they span all loading cases and the maximum of each force and moment is extracted separately. The ISRS for all four soil cases and six NPM locations were enveloped and broadened by +/-15 percent to account for uncertainty in the seismic inputs and the DB and NPM models. The broadened enveloped ISRS were later applied to the ASME Service Level D RVI FE stress analyses. The staff finds that this approach is consistent with standard practices and is acceptable. The peak spectral accelerations are clustered about lower (soft soil) and higher (hard rock) frequencies which should bound most construction locations. In-plane accelerations are amplified at low frequencies by a nonlinear sliding mechanism between the interface of the CNV skirt flange and the RXB basemat and its interaction with soil column resonance frequencies. NuScale bounds the loads from both linear (no sliding) and nonlinear (with sliding and ground resonance) analyses over all soil types to ensure conservatism. The staff finds the final bounding loads on the RVI interfaces reasonable. Finally, NuScale performed transient analyses of SG and SG support stresses using two load cases which mostly bound the upper envelope of all load cases. In particular, a soft soil and hard rock case are chosen which includes spectral peaks at low and higher frequencies. The staff finds the two load cases to be a reasonable approximation of the loading upper envelope.

3.9.2.4.5.2 Short-Term Transient Analysis of the NuScale Power Module

TR-121517, Revision 1, documents the NPM short-term transient analysis. High energy pressure boundary breaches cause short term transient events that result in an asymmetric cavity pressurization load between the CNV and RPV and blowdown loads within the RPV. The technical report contains the analytical methods, benchmarking for validating the analysis methods, and the resulting asymmetric cavity pressurization and blowdown loads. The thermal---hydraulic code NRELAP5 and the ANSYS model were used to calculate the short-term transient loads. NRELAP5 generates thermal---hydraulic boundary condition inputs for the ANSYS model, which is used to calculate the short-term- transient structural loads within the NPM, including forces, moments, and differential pressure loads. This section of the SER evaluates the ANSYS analyses of TR-121517 while SER Section 3.9.1 addresses the RELAP5 analyses.

NuScale evaluated the following valve opening and break events in TR-121517, all listed under Service Level C:

- Inadvertent opening of an RSV (saturated breach)
- Spurious RVV actuation (saturated breach)
- Spurious RRV actuation (subcooled breach)
- RCS injection line break (subcooled breach)

Eight total cases were analyzed. Different modeling parameters were used for saturated and subcooled breaches. NuScale confirmed the suitability of the modeling parameters using sensitivity and convergence studies. Hand calculations of mass flow rates at the breach locations confirmed that the chosen parameters yield reasonable results. The staff concurs that the modeling parameters are appropriate for the postulated breaks.

The ANSYS approach was benchmarked against several test cases commonly used in the nuclear reactor community; therefore, the staff finds that the benchmarking performance is reasonable. NuScale stated that no uncertainty factor is needed in the short-term transient loads analysis, citing the conservatism of the benchmarking. Preliminary ASME Service Level D stress analyses provided during the audit and in response to RAI 10111, Question 3.9.2-1, Revision 1 (see Section 3.9.2.4.5.3 of this report) show that stresses induced by seismic loads are much higher than those caused by the transient ones. Also, NuScale ignores the inadvertent actuation block feature of the RRVs which would mitigate loads from that break. Given these conservatisms and the low transient loads, the staff finds that omitting uncertainty factors is reasonable.

The CNV, RPV, RVI, and other structures are modeled with solid FEs. The fluid is modeled with solid acoustic elements. The fluid-structure interface is conformal, with coincident nodes on the structural and acoustic surfaces coupled in the normal direction. The CNV has fixed boundary conditions at the CNV lugs (circumferential direction) and CNV skirt (vertical direction at all points and radial direction at four mounting points). Flow acceleration initial conditions from the NRELAP5 analyses are applied to the acoustic elements and thrust forces applied to the solid elements. The analyses are short – 0.2 seconds long - and capture the initial transient appropriately by ensuring suitably small time-step sizes are used early in the analyses. The staff finds the ANSYS modeling and analysis approach appropriate and consistent with best practices.

To calculate blowdown loads for valve inadvertent opening in TR-121517, NuScale used the reactor coolant pressure at normal operation for the RRVs and RVVs, and the design pressure (1.1 times normal operation pressure) for the RSVs at which the RSVs are set to lift. ASME BPV Code Section III, Subsection NB requires the use of coincident pressure associated with the operating loading for the Service Level D analysis; therefore, the staff finds that the pressures selected for the blowdown analysis are appropriate.

The applicant provided bounding values of the calculated forces and moments at 123 component interface locations for all cases in Table 6-4, “Maximum Forces and Moments at Component Interfaces.” In Table 6-6, “Maximum forces and moments within component sections,” bounding values of maximum forces and moments for 30 internal sections of the NPM components such as the CNV, RPV, riser assemblies, and core barrel assembly for all break and valve opening conditions are provided. Differential pressures across the pressurizer baffle plate were also provided. The applicant stated that the highest forces and moments and differential pressures result from both RVVs opening case due to the high mass flow rate and high fluid accelerations generated in this valve opening event. The maximum forces and moments and differential pressures on RPV, CNV, and RVI due to the CVCS injection line break are bounded by the case of both RVVs opening. The NRC staff finds that the applicant considered an appropriate range of transient events and identified the most limiting transient loading conditions for the NPM.

3.9.2.4.5.3 Stress and Deflection Evaluation of Reactor Vessel Internals and Steam Generators at Service Level D Faulted Conditions

NuScale submitted a summary of the peak stress intensities of the RVI and SG at ASME service level D conditions in their response to RAI 10111, Question 3.9.2-1, Revision 1, along with detailed calculations in:

- EC-170084, Revision 0 (FE calculations for RVI using ISRS inputs),
- EC-157683, “ASME Code Qualification for Service Level D Condition of RVI Components – Classical Engineering Calculations,” Revision 1 (classical engineering calculations using peak forces and moments for RVI fasteners, welds, pins, and other small components not included in the RVI FE model), and
- EC-157339, “ASME Code Level D Evaluation of Steam Generator Tubes,” Revision 0 (Finite element calculations for SG using transient loading inputs).

NuScale included the Service Level C short term transient loads in the Service Level D evaluations. The staff examined NuScale’s detailed analyses during an in-person audit October 22-24, 2024, and in the internal calculation documents provided in response to RAI 10111, Question 3.9.2-1, Revision 1.

NuScale shows low-order modes of the NPM in TR-121515-P, Revision 1. The CNV fundamental lateral modes are those of a free beam, with the RPV moving as a rigid body cantilevered from the bottom of the CNV. The fundamental vertical modes show strong motion of the URVI with the LRVI moving in phase with the RPV. The strong URVI motion is due to the low stiffness at the URVI/LRVI interface and the bellows.

The RVI FE stress intensities were calculated using a submodel of the URVI, LRVI, and core region. The SGs were not included in the stress analysis (but the effects are included in the input loads generated from the NPM models). NuScale shows modes of the RVI assembly and of the ICIGTs in EC-170084, Revision 0 (referenced with the response to RAI 10111, Question 3.9.2-1, Revision 1). The staff examined the shapes and frequencies of the modes with the highest mass fractions and find them reasonable.

Seismic and blowdown boundary displacements were applied to all major interfaces to the RPV including at the core barrel to lower RPV interface, hanger plate to upper RPV interface, the CRDS connections, and in the lateral directions at the set screw locations. ANSYS multi-point response spectrum analysis was used to compute displacements and stress intensities. This analysis approach assumes statistically independent inputs. The staff believes statistically independent loads are unlikely for a seismic or blowdown loading which induces nearly rigid body motion of the CNV and RPV where all inputs would be correlated with each other with various relative phasing. However, it is possible that the conservatism in the bounding ISRS inputs accounts for any errors associated with ignoring relative phasing. To confirm the conservatism of the ISRS calculations, NuScale performed a full transient analysis of the RVI using a single strong transient loading case (based on the SG transient calculation experiences) to confirm the uncorrelated ISRS approach is bounding. The peak stresses from the transient calculation are within acceptable limits, but at different locations than in the ISRS calculations implying inconsistent deformations for the ISRS and transient analyses. However, because the stresses are well within acceptable limits, the staff finds the ISRS approach as implemented by NuScale (in particular the use of bounding spectra spanning all load cases and NPM locations) to be acceptable for NPM analysis.

Total deformations are dominated by the seismic response. **[[**

]]. The staff finds the deformations are small and reasonable for seismic loading events.

ASME stress intensities (the maximum difference between principal stresses) were calculated for the entire RVI submodel. The stresses induced by the seismic and blowdown loads were combined by square of the sum of squares (SRSS). Seismic induced stresses are dominant at nearly all locations. Blowdown stresses are important at the lower set screw locations but are highly localized and only occur very early in the blowdown process, with little contribution to SRSS of stresses at other locations. This means any bias errors in the blowdown loads, should they exist, are likely inconsequential for the stress analyses.

The SRSS of the stress intensities in non-ICIGT components is **[[**
]] at the CRAGT support plate. The ICIGTs have a higher maximum stress of **[[ ]]** due to their long unsupported spans. A mesh density sensitivity study confirms the stresses are reasonably converged. For conservatism NuScale compared the SRSS of the maximum stress **[[**
]].
The stress intensity generally includes both membrane and bending stresses so comparing to the low membrane limit is conservative. Final ASME stress analyses (to be delivered under ITAAC 02.01.01) will include full tables of computed and allowable stresses.

Engineering calculations of the peak stresses in bolts and other small interface components (gussets, tabs, mounting studs, threaded inserts, and welds) and simple larger structures (lower riser and core barrel cylindrical sections) were performed using the maximum forces and moments extracted from time histories of the seismic and blowdown loads. The geometries of these components are well suited to closed-form equations. Peak stresses were compared to the appropriate ASME allowables for general membrane, membrane+bending, and shear for the different component materials at 540 ^{o}F. This temperature is slightly lower than the 550 ^{o}F used for the RVI FE analyses but the staff finds it to be reasonably similar.

Stresses induced by the seismic and blowdown loads were summed by SRSS and combined with the other ASME Service Level D loads (DW – deadweight, EXT – external mechanical loads, SCR – SCRAM loads). The peak loads were provided in Table 3-4, “RVI Internals –

Forces and Moments" of EC-157683 (see response to Audit Question A-5.4.1.3-3, Table 12) and are **[[** **]]**. The staff finds that the calculated stress intensities met ASME stress allowables and small positive margins are reasonable, particularly given the conservatism in the loads used. ITAAC 02.01.01 will ensure that the ASME components will meet the ASME stress limits and are designed to withstand design-basis events. This ITAAC is evaluated in SER Section 14.3.

NuScale performed transient analyses of a FE model of the SGs including all tubes, tube supports, and backing strips. Tubes were modeled with pipe elements and the supports and backing strips with shell elements. The shell element approach was validated by comparing structural responses (static and modal analyses) of shell and solid models. The tube mesh density is higher near the bottom to capture the peak stresses more accurately. Single nodes were used to connect the tubes with the adjacent supports in all DOF, which the staff finds conservative since all transferred loading is concentrated at a single point. Tube support to tube support connections were limited to the horizonal directions. Tube material properties vary with height to account for the different temperatures of the primary and secondary coolants and the varying densities of the secondary coolant (subcooled at the bottom transitioning to steam at the top). **[[** **]]** was applied similar to that used in the loading calculations. The staff finds the SG modeling approach reasonable and consistent with best practices.

The model was analyzed for static deflections and stresses using deadweight loads to confirm that all constraints and connections were appropriate. The model was also evaluated for structural modes to ensure they are appropriate. The mode shapes and frequencies are generally consistent with those from the model used in the CVAP. The lowest modal frequencies are in the range of the peak seismic loading frequencies for hard rock conditions which the staff finds is conservative. Finally, a short time history analysis was run for this baseline model along with a model with a finer mesh. The calculated baseline model stresses are nearly identical to those in the refined model confirming the adequacy of the mesh density.

Displacement time histories from the seismic **[[** **]]** and blowdown **[[** **]]** were applied at the inlet FW plenum, set screw locations at the SG tube supports, and at the baffle plate main steam plenum. Pressure and deadweight loads were included per ASME Code requirements. The most limiting load cases were two seismic events (out of the 24 analyzed), one for hard rock and one for soft rock, both for NPM module 1; and the blowdown case with both RVVs opening (which are assumed to be the same for all modules). The staff compared the chosen seismic load cases with the upper envelope of all load cases **[[** **]]** and concurs with NuScale's choice of limiting load cases.

The seismic and blowdown stresses were combined by SRSS and added to the deadweight and pressure stresses. The final stresses are dominated by seismic loads. The peak stress intensity of 24 ksi, located near the bottom of the tubes, is less than the ASME Service level B allowable limit of 29.4 ksi. NuScale design specification requires that the Service Level D analyses of the ASME Code Class 1 components meet Service Level B limits, which are lower than the Service

Level D limits. Shear stresses are very small – about 20 times less than the allowable limit of 16 ksi.

The staff has evaluated the seismic and blowdown loading, structural analysis methodologies, and the estimated peak stresses and displacements in the RVI and SG and finds them reasonable and within allowable ASME Code limits.

3.9.2.4.6 Correlations of Reactor Vessel Internal and Steam Generator Vibration Tests with Analytical Results

Some benchmarking test data have been compared to analytic results. In particular, the forced response of the SG TF-2 was predicted using the same bounding models used by the applicant for its NPM SG design analyses. The resulting predicted response generally exceeds, sometimes significantly, TF-2 measurements, providing confidence in the conservatism of the applicant’s design analysis methods. Based on this, the NRC staff finds the TF-2 benchmarking to be acceptable, but it is not sufficient to fully validate the conservatism of NuScale’s CVAP methods. TF-3 testing, which models the NPM internal SGs, validated the adequacy of NuScale’s methods for evaluating the possibility of VS and FEI in the SGsand confirm that no VS or FEI is expected in the NPM SG.

3.9.2.5 Combined License Information Items

SDAA Part 2, Table 1.8-1, lists COL information item numbers and descriptions related to dynamic testing and analysis of SSCs from FSAR Section 3.9.2.

Table 3.9.2.-1: NuScale COL Items for FSAR Section 3.9.2

COL Item No.	Description	FSAR Section
COL 3.9-2	An applicant that references the NuScale Power Plant US460 standard design will complete an assessment of piping systems inside the Reactor Building to determine the portions of piping to be tested for vibration, thermal expansion, and dynamic effects. Piping systems within the scope of this testing include American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Class 1, 2, and 3 piping systems, other high-energy piping systems inside seismic Category I structures or those whose failure would reduce the functioning of any seismic Category I plant feature to an unacceptable level, and seismic Category I portions of moderate-energy piping systems located outside of containment. The applicant may select the portions of piping in the design for which vibration testing is performed while considering the piping system design and analysis, including the vibration screening and analysis results and scope of testing as identified by the Comprehensive Vibration Assessment Program.	3.9.2.1
COL 3.9-3	An applicant that references the NuScale Power Plant US460 standard design will verify that evaluations are performed during detailed design of the main steam lines, using acoustic	3.9.2.1.1.1

	resonance screening criteria and additional calculations as necessary (e.g., Strouhal number) to determine if there is a concern. The methodology in “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report,” TR-121353 is acceptable for this purpose. The applicant will update Section 3.9.2.1.1.1 to describe the results of this evaluation.	
COL 3.9-4	An applicant that references the NuScale Power Plant US460 standard design will provide applicable test procedures before the start of testing and will submit test and inspection results from the Comprehensive Vibration Assessment Program for the NuScale Power Module in accordance with Regulatory Guide 1.20.	3.9.2.4

3.9.2.6 Conclusion

Pending verification of the references in FSAR Section 3.9.2.1 regarding the FSAR Section 14.2 initial startup piping vibration and thermal expansion testing, the NRC staff concludes that by having an acceptable vibration, thermal expansion, and dynamic effects test program that will be conducted during the initial startup on specified high- and moderate-energy piping, and all associated systems, restraints, and supports, the design will meet the relevant requirements of 10 CFR 50.55a and GDC 1, 2, 4, 14, and 15. These tests provide confirmation that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions encountered during service and that adequate clearances exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations.

The NRC staff finds that the applicant’s FIV analysis and testing procedures are reasonable and conservative. Important FIV mechanisms have sufficient margin to provide reasonable assurance against their occurrence during normal and faulted NPM operation. TF-3 test data confirmed the safety of the SG for FIV and VS and FEI in particular. NPM initial startup testing will capture any unexpectedly high FIV using arrays of externally mounted pressure sensors. Initial startup testing will also confirm the lack of significant ARs in the CNTS steam piping. The applicant has committed to performing all testing in compliance with Criterion XI of 10 CFR Part 50, Appendix B.

Full inspection following the initial startup testing, followed by periodic inspections throughout the life of the plant, including the SGs, provide further confidence in the safety of the RVIs and SG. The NRC staff concludes there is reasonable assurance that there will be no significant RVI degradation due to FIV during the life of an NPM. The NRC staff also concludes that the NuScale design meets the requirements of 10 CFR Part 50, Appendix A, GDC 1 and 4, for design and testing of reactor internals to quality standards commensurate with the importance of the safety functions performed with appropriate protection against dynamic effects, including FIV and AR. The NRC staff also concludes that the CVAP for the first reactor module, in accordance with the regulatory positions of RG 1.20, provides an acceptable basis for design adequacy of the reactor internals under test loading conditions comparable to those experienced during operation without significant secondary coolant DWO instabilities. Finally, the NRC staff concludes that the design will meet the relevant requirements of Appendix B to 10 CFR Part 50 and GDC 1 and 4, with regard to the internals of a prototype reactor being tested to quality standards commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects.

The staff has evaluated the possibility of severe DWO occurring in the NPM and finds it highly unlikely that DWO-induced loads at and around the SG inlets, such as those due to CIWH and cavitation, will occur. NuScale's approach temperature limits will ensure that the **[[ ]]**, precluding these loading types. In the highly unlikely event repeated severe DWO occurs, NuScale has confirmed that CIWH and cavitation events will not lead to significant SG damage. Finally, the inspection program ensures all tubes are inspected after the first refueling outage and after 72 effective full power months of plant operation afterwards. Any tubes violating wear limits will be plugged.

The NRC staff concludes that appropriate dynamic system analyses have been performed to confirm that the structural design of the reactor internals is able to withstand the dynamic loadings of the most severe short term transient events in combination with SSE, and other loads with no loss of function. The NRC staff also concludes that the methods and procedures for dynamic systems analyses, the considerations in defining the mathematical models, the descriptions of the acceptance criteria, and the interpretation of the analytical results comply with the relevant requirements of Appendix S to 10 CFR Part 50 and GDC 2 and 4.