3 DESIGN STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

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

3.2 Classification of Structures, Systems, and Structures

3.3 Wind and Tornado Loading

3.4 Water Level (Flood) Design

3.5 Missile Protection

3.6 Protection against Dynamic Effects Associated with Postulated Rupture of Piping

3.7 Seismic Design

3.8 Design of Category I Structures

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

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

3.9.2.1 Introduction

This section of the safety evaluation report (SER) reviews the analytical methodologies, testing procedures, and dynamic analyses that the applicant used to ensure the structural and functional integrity of the piping systems, mechanical equipment, reactor vessel internals (RVIs), and their supports under vibratory loadings, including those caused by fluid flow and postulated seismic events.

This section addresses six main areas of review:

(1) piping vibration, thermal expansion, and dynamic effects testing;

(2) seismic analysis and qualification of seismic Category I mechanical equipment;

(3) dynamic response analysis for RVIs under operational flow transients and steady-state conditions;

(4) preoperational flow-induced vibration (FIV) testing of RVI;

(5) dynamic system analysis of the RVIs under faulted conditions; and

(6) correlations of RVI vibration tests with the analytical results.

3.9.2.2 Summary of Application

Design Certification Application (DCA) Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1, "NuScale Power Module," provides the information associated with this section on designing the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PV Code) Class core support components in the NuScale Power Module (NPM) to the provisions of ASME B&PV Code, Section III, Subsection NG, "Core Support Structures."

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," describes the dynamic testing and analysis of systems, components, and equipment. DCA Part 2, Tier 2, Section 3.9.2.1, "Piping Vibration, Thermal Expansion, and Dynamic Effects," and Section 14.2, "Initial Plant Test Program," describe the vibration and thermal expansion testing of piping.

DCA Part 2, Tier 2, Section 3.9.2.5, "Dynamic System Analysis of the Reactor Internals under Service Level D Conditions," and DCA Part 2, Tier 2, Appendix 3A, "Dynamic Structural Analysis of the NuScale Power Module," describe the dynamic analysis of the NPM. DCA Part 2, Tier 2, Appendix 3A, references Technical Report (TR)-0916-51502, "NuScale Power Module Seismic Analysis," Revision 1, issued September 2018 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18271A193 (proprietary) and ML18271A191 (nonproprietary)), for additional details. A system-level NPM seismic analysis was performed first on a three-dimensional (3-D) 360-degree NPM ANSYS model that consists of five NPM submodels representing the containment, reactor vessel, lower RVI, upper RVI, and control rod drive mechanism (CRDM). The five submodels were connected by coupling nodes, constraint equations, contact elements, and fluid coupling to form the 3-D NPM model. A soil-structure interaction (SSI) analysis was performed using a simplified NPM model with a peak ground motion of 0.5g (gravitational acceleration) in the horizontal direction and 0.4g in the vertical direction. The calculated acceleration time histories from the SSI analysis were used as boundary conditions at the pool floor and walls, as well as the NPM supports for seismic analysis of the NPM entire pool model that contains the 3-D NPM and entire pool water. The six analysis cases considered cracked and uncracked concrete of the reactor building (RXB), variation of NPM stiffness, and location of the NPM. From the results of the six cases, maximum seismic loads (i.e., displacements, in-structure response spectra (ISRS), forces and moments at NPM component interfaces, and cross sections) were generated for the NPM component-level stress analysis in the Service Level D Condition. The load combination in the Service Level D Condition component stress analysis involved the safe-shutdown earthquake (SSE) load, blowdown load of the main steam pipe break (MSPB) and feedwater pipe break (FWPB), and design-basis pipe break (DBPB) load. The DBPB load included the load from the spurious actuation of the reactor vent valves (RVV), reactor safety valves (RSV), and reactor recirculation valves (RRV) and from a chemical and volume control system (CVCS) pipe break. The square root of the sum of squares (SRSS) method was used to combine the loads resulting from an SSE and the highest of the MSPB, FWPB, or DBPB. TR-1016-51669, "NuScale Power Module Short-Term Transient Analysis," Revision 0, issued December 31, 2016 (ADAMS Accession Nos. ML17005A154 (proprietary) and ML17005A132 (nonproprietary)), documents the blowdown analysis of a DBPB, MSPB, and FWPB. The thermal-hydraulic code NRELAP5 and the ANSYS code calculated the short-term transient structural loads. NRELAP5 was used to generate thermal-hydraulic boundary condition inputs for the ANSYS structural dynamic analysis model that calculated the short-term transient structural loads within the NPM. Two ANSYS models were developed for (1) calculating the asymmetric cavity pressurization load between the containment vessel (CNV) and the reactor pressure vessel (RPV) and (2) calculating the blowdown load inside the RPV. For DBPB, seven cases were analyzed.

Maximum forces and moments at NPM component interfaces and cross sections were generated for each of the seven analysis cases. The bounding values of the maximum forces and moments of the seven analysis cases were used to combine with the SSE load for the NPM component Service Level D stress evaluation.

NuScale has submitted TR-0716-50439, "NuScale Comprehensive Vibration Assessment Program Technical Report," Revision 1, issued January 2018 (ADAMS Accession Nos. ML18022A223 (proprietary) and ML18022A221 (nonproprietary) respectively), which is referenced in DCA Part 2 Tier 2, Section 3.9.2.3, "Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions," and Section 3.9.2.4, "Flow-Induced Vibration Testing of Reactor Internals before Unit Operation." In addition, NuScale has submitted licensing document TR-0918-60894, "NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report," Revision 0, issued December 2018 (ADAMS Accession Nos. ML18344A029 (proprietary) and ML18341A337 (nonproprietary)).

The NPM comprises a reactor core, pressurizer, and two helical coil steam generators (HCSGs) within a cylindrical RPV, which is housed in a cylindrical steel CNV. The NPM operates passively with primary coolant circulating naturally and slowly, reducing the strength of flow-induced forces compared to those in a typical PWR. The NPM has no pumps; therefore, there are no pump dynamic forces or inlet flow jets to impinge on reactor components. In addition, all primary coolant is in a fully liquid state to ensure that two-phase flow does not occur. The NPM rests in a reactor pool that acts as a heat sink and allows for passive operation (i.e., pumps are not used to circulate or inject coolant) and passive safety systems (i.e., decay heat removal system (DHRS) and emergency core cooling system (ECCS)). A power plant comprises up to a maximum of 12 NPMs.

The NuScale RVI is a first-of-its-kind design. For that reason, it is classified as a prototype. Following the NuScale RVI's qualification as a valid prototype, future NPMs will be considered Category I nonprototypes. A single NPM is smaller than currently operating pressurized-water reactors (PWRs) and produces a power output of 50 megawatts electric. Unlike traditional PWRs with forced primary coolant circulation, the core flow rate is proportional to plant power. The HCSG is integral to the NPM and, therefore, is evaluated along with reactor vessel internals (RVIs) as well as the steamlines and primary coolant piping up to the NPM disconnect flange, outboard of the isolation valves in each line for FIV effects. FIV effects on the RVIs, HCSG, and steamlines/valves are evaluated during normal operation, decay heat removal (DHR), and emergency core cooling conditions. Although the RVIs will experience worst case FIV loads during normal operation, the steamlines and isolation valves experience FIV loads during ECCS or DHRS operation.

The current comprehensive vibration assessment program (CVAP) describes the screening and provides the results of the FIV analyses of (1) RVI and structures, (2) HCSG components, and (3) reactor coolant system (RCS) piping, up to the NPM disconnect flange and including the isolation valves.

Components with small margins of safety against FIV effects were identified for validation testing. In particular, an HCSG mockup, called SIET test facility- 3 (TF-3), will be built and tested in accordance with TR-0918-60894. In addition, a final validation test for leakage flow instability (LFI) of the HCSG inlet flow restrictor is planned. The initial startup testing for FIV effects on the prototype NPM will be limited to a modest set of sensors to detect any unexpectedly strong FIV of the RVIs and HCSG. In addition, DHRS and main steam piping will be instrumented during initial startup testing to ensure that there are no flow-excited acoustic resonances (ARs).

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): DCA Part 2, Tier 1, Table 2.8-2, “Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria,” ITAAC No. 1, requires that the seismic Category I equipment listed in DCA Part 2, Tier 1, Table 2.8-1, “Module Specific Mechanical and Electrical/I&C Equipment,” including its associated supports and anchorages, withstands design-basis seismic loads without the loss of its safety-related function(s) during and after an SSE. DCA Part 2, Tier 1, Table 2.8-1, lists seismic Category I RVI components.

Technical Specifications: None

Technical Reports:

- TR-0716-50439
- TR-0918-60894
- TR-0916-51502
- TR-1016-51669

3.9.2.3 Regulatory Basis

The following relevant U.S. Nuclear Regulatory Commission (NRC) regulatory requirements apply to this review:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, “Codes and Standards,” as it relates to the design, fabrication, erection, and testing of structures, systems, and components (SSCs) in accordance with the quality standards that are commensurate with the importance of the safety function to be performed.

- General Design Criterion (GDC) 1, “Quality Standards and Records,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” 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.

- 10 CFR Part 50, Appendix A, GDC 2, “Design Bases for Protection against Natural Phenomena,” 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.

- 10 CFR Part 50, Appendix A, GDC 4, “Environmental and Dynamic Effects Design Bases,” 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.

- 10 CFR Part 50, Appendix A, GDC 14, “Reactor Coolant Pressure Boundary,” as it relates to designing SSCs of the reactor coolant pressure boundary to have an extremely low probability of abnormal leakage, rapidly propagating failure and gross rupture.

- 10 CFR Part 50, Appendix A, GDC 15, “Reactor Coolant System Design,” as it relates to designing the RCS with sufficient margin to assure that the reactor coolant pressure boundary is not exceeded during normal operating conditions, including anticipated operational occurrences.

- 10 CFR 52.47(b)(1), which requires a DCA to include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification (DC) is built and will operate in accordance with the DC; the provisions of the Atomic Energy Act of 1954, as amended; and NRC regulations.

- Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50, as it relates to the quality assurance criteria for the dynamic testing and analysis of SSCs.

- Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” to 10 CFR Part 50, as it relates to certain SSCs that must be designed to remain functional for an SSE.

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 3.9.2, “Dynamic Testing and Analysis of Systems, Structures, and Components,” lists the acceptance criteria adequate to meet the above requirements and review interfaces with other SRP sections. In addition, the following guidance documents provide acceptance criteria that confirm that the above requirements have been adequately addressed:

- Regulatory Guide (RG) 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing,” Revision 3, issued March 2007, as it relates to the vibration analysis and testing methodologies of the RVIs.

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

- RG 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” Revision 1, issued February 1978, as it relates to the development of floor response spectra.

- 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 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.

3.9.2.4 Technical Evaluation

3.9.2.4.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

DCA Part 2, Tier 2, Section 3.9.2.1, addresses the initial startup testing that is performed to verify that the vibrations and thermal expansion and contraction of the as-built piping systems are bounded by the design requirements. The piping systems in the initial startup testing program include (1) ASME B&PV Code, Section III, Class 1, 2, and 3 piping systems identified in DCA Part 2, Tier 2, Table 3.2-1, “Classification of Structures, Systems, and Components,” (2) other high-energy piping systems inside seismic Category 1 structures or those whose failure would reduce the functioning of any seismic Category I plant feature to an unacceptably level, and (3) seismic Category I portions of moderate-energy piping systems located outside of containment.

DCA Part 2, Tier 2, Table 3.2-1, lists systems and their quality group that correspond to the ASME B&PV Code classification. The test program, as described in DCA Part 2, Tier 2, Section 14.2, verifies that the Class 1, Class 2, Class 3, and other high-energy and seismic Category 1 piping systems meet functional design requirements and that piping vibrations and thermal expansions are within acceptable levels and will withstand dynamic effects resulting from operating transients.

DCA Part 2, Tier 2, Section 3.9.2.1, states that DCA Part 2, Tier 2, Section 14.2, describes the initial test program. The vibration, thermal expansion, and dynamic effect elements of this test program are performed during Phase I preoperational testing and Phase II initial startup testing. The Phase I preoperational tests are performed to demonstrate that the piping system components meet functional design requirements and that piping vibrations and thermal expansions and contractions are bounded by the analyses. If the design-basis parameters are not bounding compared to the measured values, corrective actions (i.e., reanalyzing with as-built values) are implemented, and the systems are retested. The Phase II initial startup testing is performed after the reactor core is loaded into a reactor module. These Phase II tests determine that the vibration level and piping reactions to transient conditions are acceptable and are bounded by the analyses. If the vibration levels are not bounded, the analyses use the vibration level from the testing as input to subsequently verify that the design is acceptable.

DCA Part 2, Tier 2, Section 3.9.2.1.1, “Piping Vibration Details,” states that vibration test specifications were developed and that piping vibration testing and assessment were performed in accordance with ASME Operation and Maintenance of Nuclear Power Plants, Division 2, 2012 Edition, Part 3. SRP Section 3.9.2, Revision 3, accepted the ASME OM Standards and Guides 2000 Edition. The NRC staff finds that the use of ASME OM Code, Division 2, 2012 Edition, is acceptable because the provisions for piping vibration and thermal expansion testing are equivalent.

DCA Part 2, Tier 2, Section 3.9.2.1, includes Combined License (COL) Item 3.9-13 for the COL applicant to assess and select piping systems for the vibration and thermal expansion testing using the piping vibration screening and analysis results of the CVAP. The staff finds that the COL item adequately addresses the assessment and selection of the piping system for vibration testing during initial startup testing. Additionally, ASME OM Code, Division 2, 2012 Edition, Part 3, does not specify the criteria for selecting piping for vibration testing; therefore, considering the screening and analysis results of the CVAP for the selection of piping systems for vibration testing is an acceptable approach.

DCA Part 2, Tier 2, Section 3.9.2.1.1.1, "Main Steam Line Branch Piping Acoustic Resonance," addresses the concern of main steamline (MS) branch piping ARs. COL Item 3.9-10 addresses the detailed design of the MS piping by the COL applicant. This COL item ensures that the detailed design of the MS line considers the phenomenon of AR and the piping vibration screening and analysis results of the CVAP. The staff finds the COL item and the process used to complete the detailed design the MS line to avoid AR acceptable because COL Item 3.9-10 will ensure that the design of the piping systems will preclude vibration caused by ARs at closed pipe branches.

DCA Part 2, Tier 2, Section 3.9.2.1.2, "Piping Thermal Expansion Details," states that thermal expansion testing verifies that the design of the piping systems tested prevents constrained thermal contraction and expansion during Service Level A and B transient events. In addition, the tests verify that the component supports can accommodate the expansion of the piping for the service levels for these modes of operation. DCA Part 2, Tier 2, Section 14.2, describes selected planned piping thermal expansion measurement tests. Test specifications for thermal expansion testing of piping systems during preoperational and startup testing will be done in accordance with ASME OM Code, Division 3, 2012 Edition, Part 7. The staff finds that performing the piping thermal expansion testing according to OM Code, Division 3, 2012 Edition, Part 7, is acceptable because this meets the SRP guidance. The initial startup testing provides adequate assurance that the piping and piping restraints of the system are designed to withstand thermal effects during normal and transient operating conditions.

3.9.2.4.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

DCA Part 2, Tier 2, Section 3.9.2.2, "Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment," references DCA Part 2, Tier 2, Section 3.7, "Seismic Design"; Section 3.10, "Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment"; and Section 3.12, "ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports." The corresponding sections of this SER include the review of these sections.

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

Design and Operation Summary

DCA Part 2, Tier 2, Section 1.2.1.1.2, "Operating Characteristics," and Section 3.9.5, "Reactor Vessel Internals," describe the NPM and its RVI components. Aspects of the design relevant to FIV are summarized here. The RPV is mounted within the steel CNV, which operates in a large pool of water. The CNV also contains auxiliary piping, including the CVCS and piping connection to the DHRS. The RPV is a 58-foot–high (17.7 meters), 9-foot–diameter (2.7 meters) cylinder that is rated to operate at 1,850 pounds per square inch, absolute (psia) (12.8 Megapascal (MPa)). 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 penetrations (less than 8 inches in diameter) accommodate the pressurizer spray, RVVs, RSVs, and CRDMs. The CRDMs are mounted on top of the RPV with rods extending downward into the RPV.

The core support assembly and hot-leg riser system comprises the core support structure and a lower riser assembly that rests on the core support assembly. The upper riser assembly is suspended from the pressurizer baffle plate by an upper hanger brace and sits over the lower riser with a slip joint along a conical mating section. A bellows is included in the upper riser to allow for relative movement between the upper and lower risers (primarily because of thermal expansion) at the slip joint and to minimize the likelihood of significant leakage flow between the hot and cold legs.

The dome of the RPV houses the pressurizer system. The primary coolant turns downward below the pressurizer baffle plate, passes the steam generator (SG) tubing in the outer annulus of the RPV. Pressure is regulated by a pair of heater bundles, which may be activated to increase pressure, and two spray nozzles connected to the CVCS, which provide subcooled water at the top of the 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. Support plates with several holes are attached to the risers to accommodate control rod drive (CRD) shafts and in-core instrumentation guide tubes (ICIGTs), which are inserted into the top of the RPV and extend downward into the core to monitor and control the reactor. The CRD shafts can move upward and downward, whereas the ICIGTs are stationary. Nominal clearances are specified between the hole boundaries in the grillages (called CRD shaft support plates) and the CRD shaft and ICIGT structures.

The once-through HCSGs consist of two independent arrays of tubing within the annulus between the riser and outer wall of the RPV. The tubing for each array (or bundle) is welded to tube sheets at two integral feed and steam (near the top) plenums, thus forming a pressure boundary between the primary and secondary coolant. The tubing is held in place by arrays of tube support assemblies mounted on upper SG supports attached to the pressurizer baffle plate and interfaced with lower SG supports that are also attached to the RPV. Nominal clearances are specified between the tubing supports and tubes. Inlet flow restrictors are mounted to a plate and inserted into all HCSG tubing inlets to prevent low frequency density wave oscillation from occurring within the tubes during normal operation.

A CVCS purifies the primary coolant as needed. CVCS piping protrudes through the RPV walls into the annulus between the RPV wall and upper riser assembly above the core exit. The pressurizer nozzles add primary coolant, as necessary, to reduce pressure in the pressurizer region at the top of the RPV. Flow through the CVCS is very slow and not likely to induce significant forces on the RVIs.

The reactor operates passively, with primary coolant flowing upward through the core and the lower and upper riser assemblies, then moving radially outward along the top of the RPV and then downward though the annulus between the risers and RPV wall over the SG tubing array. All flow is single phase (i.e., no boiling occurs within the primary coolant region). After passing over the HCSG tubing and through the downcomer, the flow moves radially inward before proceeding upward through the core and riser assemblies again. The secondary coolant enters the bottom of the HCSG tubing as preheated liquid and travels upward opposite the primary coolant flow direction. As heat is transferred from the primary to the secondary coolant, the secondary coolant within the HCSG tubing boils and transitions to superheated steam, which then exits into a plenum and steam supply nozzles near the top of the RPV where it travels to the steam turbines. Because the primary flow is passive, it is about 5 to 25 times slower than flow in a traditional PWR. However, the NuScale internal components are also much smaller than those in traditional PWRs; therefore, FIV still needs to be assessed.

The HCSG tubing functions in conjunction with the DHRS. The DHRS provides secondary-side reactor cooling for non-loss-of-coolant accidents when normal feedwater is not available. For DHRS operation, the main feedwater and MS isolation valves are closed, and the DHRS valves are opened. Water/steam in the secondary loop circulates naturally through the DHRS in the reactor pool and HCSG loops inside the reactor. Condensers are connected to the two SG loops, rejecting heat to the water in the reactor pool. The flow rates through the HCSG during DHRS operation are lower than those during normal operation; therefore, DHRS operating conditions do not need HCSG FIV evaluation. However, flow over cavities and standpipes in the DHRS piping is evaluated for AR.

The ECCS provides primary side cooling and coolant inventory control for loss-of-coolant accidents. For ECCS operation, and two sets of emergency core cooling valves are opened. The RVVs release the primary coolant in the RPV to the CNV, where it condenses on the inner walls. The RRVs located above the core also open to allow natural circulation between the condensed water in the annulus between the CNV and the RPV and the water within the RPV. Because flow rates throughout the ECCS process are low, FIV loads are small. In addition, because the duration of initial transients is short, any alternating stresses that are induced do not occur for many cycles.

<u>Analytic Flow-Induced Vibration Evaluation</u>

The staff based this evaluation of the applicant's FIV RVI and HCSG analyses on the NuScale CVAP and two audits of NuScale's internal documents, drawings, and test data that were conducted from May 16, 2017, through November 2, 2107, and September 5, 2018, through October 4, 2018. The staff used the audits to assess the details of the applicant's analyses. Detailed audit reports are available in ADAMS under Accession Nos. ML18023A091 (nonproprietary)/ML18022A377 (proprietary) and ML18333A221 (nonproprietary)/ML18333A222 (proprietary); the SER presents only significant aspects of the audits.

The applicant screened the following components for FIV:

- RVIs and structures;
- HCSG components; and
- Primary and secondary coolant piping up to the NPM disconnect flanges including the isolation valves.

The applicant initially screened all components and evaluated those components that it deemed susceptible to any of the FIV mechanisms in more detail. Based on the evaluations, the applicant identified selected components for testing to confirm the analyses. The staff finds the screening procedures acceptable because they are consistent with the guidance in ASME B&PV Code, Appendix N, and the open literature. The screening procedures identified components that need more detailed FIV analyses. The applicant evaluated the following components for FIV effects resulting from primary coolant flow:

- SGs
    - tubing
    - tube support bars
    - lower tube support cantilevers
- upper riser assembly

- upper riser section
- riser slip joint
- ICIGTs
- CRD shaft
- CRD shaft support
- upper riser hanger brace

- lower riser assembly
  - lower riser section
  - control rod assembly guide tubes (CRAGTs)
  - upper core plate
- core support assembly
  - core barrel
  - upper support block
  - core support block
  - reflector block
  - lower core plate
  - fuel pin interfaces
- other RVIs
  - pressurizer spray system
  - CVCS injection system
  - flow diverter
  - thermowells
  - component and instrument ports

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

- steam piping, nozzles, and MS isolation valves;
- HCSG steam plenum;
- DHRS steam and feedwater piping;

- DHRS condensate piping;
- HCSG tubing; and
- steam generator tube inlet flow restrictors (SGIFRs).

The staff usually does not review SGs as part of an RVI CVAP. However, because the HCSG tubing is integral to the NuScale reactor module, the staff reviewed it for FIV as part of the RVI CVAP. The staff is concerned that limited margins against vortex shedding, turbulent buffeting, and fluid-elastic instability exists in the FIV analysis for the HCSG tubing, and are based on potentially non-conservative modeling and assumptions. The staff issued **Request for Additional Information (RAI) 427-9408, Question 03.09.02-74** (ADAMS Accession No. ML18107A802), asking NuScale to address these concerns. The staff is tracking **RAI 427-9408, Question 03.09.02-74,** as **Open Item 03.09.02-1**.

The staff finds that the components evaluated for FIV are reasonable and that it is unlikely that any other components are susceptible to FIV based on low-flow conditions or robust structural designs, or both.

The applicant addressed the following FIV mechanisms:

- turbulent buffeting (TB);
- flutter and galloping (F/G);
- LFI;
- vortex shedding (VS);
- fluid-elastic instability (FEI); and
- AR.

These are the usual FIV mechanisms evaluated in a CVAP, and the staff finds them acceptable (no other possible FIV mechanisms are neglected). For TB, NuScale also assessed fatigue life and wear associated with contact between adjacent components. All other FIV mechanisms are only evaluated for their presence because they are associated with the "lock-in" of structural or acoustic motion with a flow-induced excitation mechanism or instability. If sufficient margin against the possibility of lock-in exists, the response to flow-induced loads is assumed to be negligible. However, because NuScale's margins against lock-in of structural resonances with VS loads are small, this assumption may be inappropriate, and forced response calculations for VS may be needed, as further discussed below.

NuScale relies heavily on ASME B&PV Code, Section III, Nonmandatory Appendix N-1300, "Flow-Induced Vibration of Tubes and Tube Banks," for much of its screening; a workbook by M.K. Au-Yang, "Flow-Induced Vibration of Power and Process Plant Components: A Practical Workbook," issued in 2001; and a paper by S.S. Chen on FEI and VS in HCSG tubing (see S.S. Chen, "Tube Vibration in a Half-Scale Sector Model of a Helical Steam Generator," *Journal of Sound and Vibration* 91(4):539–569, 1983). Unlike previous applicants that have submitted comprehensive scale models or full-scale plant test data, or both, to substantiate their analysis procedures, NuScale has performed minimal benchmarking. The staff assessed this lack of benchmarking during the CVAP review and the 2017 and 2018 audits (ADAMS Accession Nos. ML18023A091 and ML18333A221, respectively), as further discussed below.

An FIV mechanism was evaluated for a given component using a combination of the following:

- forcing function methodologies (from the ASME B&PV 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 finite element (FE) analyses); and
- assumed structural damping.

The ANSYS software suite, which includes structural FE and CFD modeling tools, was used to support the FIV analyses of RVI and the HCSG. ANSYS is a preverified and configuration-managed FE analysis program used in the design and analysis of safety-related components. It was used for structural modal analysis and CFD primary coolant flow simulations. Although the software is acceptable, the staff finds that meshing procedures and densities, boundary condition assumptions, and fluid loading effects should be appropriate and conservative for a given model to be considered reasonable and bounding. The staff issued **RAI 386-9316, Question 03.09.02-52** (ADAMS Accession No. ML18072A149), asking the applicant to address the staff's concerns associated with validating the FE modeling procedures. The staff later issued **RAI 427-9408, Question 03.02.02-74,** which included questions about other possible nonconservative aspects of NuScale's FIV analyses and included the staff's concerns raised in **RAI 386-9316, Question 03.09.02-52**. The staff is tracking **RAI 386-9316, Question 03.09.02-52,** and **RAI 427-9408, Question 03.02.02-74,** as **Open Item 03.09.02-2 and Open Item 03.09.02-1,** respectively.

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

The staff has two concerns with the velocities NuScale used in its FIV analyses:

- The empirical TB forcing models assumed simple canonical flow fields (e.g., annulus or pipe flows) and were scaled by the peak velocity, usually at the center of a flow profile.

Using the average velocity across a flow profile was not consistent with the forcing model assumptions and biased the forcing amplitudes nonconservatively. The staff issued **RAI 427-9408, Question 03.09.02-73** (ICIGT and CRDS FIV), and **Question 03.09.02-74** (HCSG FIV) (ADAMS Accession No. ML18072A149), asking NuScale to address this concern. The staff is tracking these RAIs as **Open Item 03.09.02-3** and **Open Item 03.09.02-1,** respectively.

- The flow velocities across the HCSG tubing were assumed to be uniform. Uniform flow is highly unlikely because of the pressure and temperature gradients throughout the HCSG region. Localized velocity deficits and increases are likely, with perhaps as much as 20- to 30-percent variability based on HCSG studies in the open literature. Faster flows will cause higher VS frequencies, which could lock-in to structural tube resonances, and an increased possibility of FEI. The staff issued **RAI 427-9408, Question 03.09.02-74** (ADAMS Accession No. ML18072A149), asking NuScale to address this concern. The staff is tracking **RAI 427-9408, Question 03.09.02-74,** as **Open Item 03.09.02-1**.

In addition, the staff had concerns about the empirical forcing functions used to assess FIVs of the CRAGTs. NuScale's FIV analysts stated in several locations in their internal documents that the CRAGT flow is highly complex and that forcing functions are unknown (see the 2017 audit report (ADAMS Accession No. ML18023A091)). The staff's review of the CRAGT geometry and the core region confirms that the flow will pass both within and outside the CRAGTs, with flow possibly repeatedly entering and exiting the CRAGTs through many holes in the sides of the tubes. This sort of flow is not well understood, and no bounding forcing models are available in the ASME B&PV Code or open literature. Early versions of NuScale's internal documents stated that test data would be used to quantify the forcing functions. Later revisions of the documents removed those planned tests without justification. NuScale's original TB fatigue damage analysis of the CRAGTs estimated that over [ ] of the wall thickness could be worn away over the life of the plant. In a letter dated July 25, 2018 (ADAMS Accession No. ML18206A815), NuScale provided two means of establishing the long-term structural integrity of the CRAGTs. First, similar designs in other plants have been operating for many years under much higher flow speeds with no observed degradation. Second, the original NuScale wear analysis was shown to be excessively conservative. More realistic (but still conservative) analyses reduced the estimated wear to [ ] of wall thickness, which is very low. Finally, in accordance with TR-0918-60894, Table 7-1, "Pre- and Post-Initial Startup Testing Inspection Locations," the CRAGT was included in the inservice inspection program to ensure that unforeseen degradation does not progress to the point of component failure over the life of the plant. The staff finds the clarification on the CRAGT wear analysis acceptable because the inservice reliability of similar designs in stronger flow fields in operating plants and revised (but still conservative) analysis results provide reasonable assurance that FIV will not adversely affect the NuScale CRAGTs.

Although the primary coolant flow is the main source of FIV in the NuScale reactor, turbulent secondary coolant flow will also drive the inner walls of the HCSG tubing. NuScale used simple turbulent pipe flow models for these forces, but it also conducted separate effects testing of the wall pressures in SIET TF-1. The CVAP did not include these TF-1 test data; however, the staff examined them during the 2017 and 2018 audits. TR-0918-60894 describes some of these data. Strong spectral peaks were observed in the measured wall pressure data; however, NuScale did not explain these peaks nor use them in the FIV analyses of the HCSG tubing. The secondary coolant enters the HCSG tubing as preheated water, transitions to boiling on its way to the steam headers and exits as superheated steam. Measurements in the open literature clearly show strong spectral peaks in two-phase (boiling) internal upward flow within tubing. The staff issued **427-RAI 9408, Question 03.09.02-74** (ADAMS Accession No. ML18107A802**)**, asking NuScale to address the effects of these forces on HCSG tubing fatigue damage in its analyses. The staff is tracking **RAI 427-9408, Question 03.09.02-74,** as **Open Item 03.09.02-1**.

NuScale 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 the concerns raised above about average velocities do not challenge the margins; therefore, the staff finds the F/G analyses acceptable.

The applicant used ANSYS FE analyses to estimate structural mode shapes and resonance frequencies. External (primary coolant) fluid mass loading was not modeled explicitly; instead, it was assumed to be that of the volume displaced by a given structure, which is a reasonable bounding approximation per ASME B&PV Code Appendix N. The HCSG tubing was assumed to be completely full of water (secondary coolant) to compute the lowest possible resonance frequencies, which is also bounding. All the FE models are of individual components, with assumed idealized boundary conditions. No models were constructed of overall assemblies. Based on images of the FE models in the NuScale internal documents, the meshes are all coarse and no studies were performed to ensure that simulated resonance frequencies converged (coarse FE models bias resonance frequencies high). In addition, the models of the ICIGTs, CRD shafts, and HCSG tubing assumed zero translational motion boundary conditions at nearby supports, which bias the lowest resonance frequencies higher. The ICIGTs and CRD shafts pass through holes in the CRD support plates. However, the clearances between these structures and the holes did not support the assumption of fixed boundaries. In addition, the clearances specified between the HCSG tubing and tube supports were inconsistent with NuScale's assumed fixed in-plane (radial) boundary conditions. Actual operational clearances may be different because of tubing weight and hydrodynamic or thermal effects, or both, but these operational clearances have not been quantified and were not made available to the staff. The coarse meshes and idealized boundary conditions may have caused the NuScale resonance frequency estimates to be biased high. Because the most critical FIV analyses were of possible lock-in of structural resonances with VS and FEI, comparing the VS and FEI frequencies with resonance frequencies that are biased high is nonconservative. The staff issued **RAI 427-9408, Question 03.09.02-73** and **Question 03.09.02-74** (ADAMS Accession No. ML18107A802), asking NuScale to address the effects of the lack of mesh convergence studies and assumed idealized boundary conditions on the ICIGTs, CRD shaft, and HCSG FIV analyses. The staff is tracking **RAI 417-9408, Question 03.09.02-73** and **Question 03.09.02-74,** as **Open Item 03.09.02-3** and **Open Item 03.09.02-1,** respectively.

Damping of all RVIs is assumed to be less than or equal to 1 percent, which is acceptable in accordance with RG 1.20. However, NuScale assumed 1.5-percent damping for the HCSG tubing VS and FEI analysis but did not provide test data to substantiate this increased damping. The higher damping is assumed to be caused by interaction between the tubing and tube supports, which depends on the tightness of fit that, in turn, depends on thermal expansion of the tubing and tube supports at operating conditions. The higher assumed damping leads to higher estimated margins against VS and FEI. NuScale stated during the 2017 and 2018 audits (ADAMS Accession Nos. ML18023A091 and ML18333A221, respectively) that upcoming SIET TF-3 testing (as described in TR-0918-60894) will substantiate the higher damping. The staff issued **RAI 9408, Question 03.09.02-74** (ADAMS Accession No. ML18107A802), to ask NuScale to provide the SIET TF-3 test plans and a quantitative assessment of the tightness of fit between the HCSG tubing and supports and to explain how that would be simulated in the TF-3 facility. The staff is tracking **RAI 427-9408, Question 03.09.02-74,** as **Open Item 03.09.02-1**.

As discussed in the 2017 audit report (ADAMS Accession No. ML18023A091), NuScale's internal documents describe the standard random analysis methods typically used to calculate structural response resulting from TB loads; however, they also suggest that the TB forced response analyses for some components are performed with approximate random analysis methods proposed by B. Brenneman (see B. Brenneman, "Random Vibrations Due to Small-Scale Turbulence with the Coherence Integral Method," *ASME Journal of Vibration, Stress, and Reliability in Design*, 109(2):158–161, April 1987). It is unclear which components the Brenneman methods were applied to, or that the applicant applied them conservatively and appropriately. The staff issued **RAI 427-9408, Question 03.09.02-73** and **Question 03.09.02-74** (ADAMS Accession No. ML18107A802), asking NuScale to address these concerns. The staff is tracking **RAI 427-9408, Question 03.09.02-73** and **Question 03.09.02-74,** as **Open Item 03.09.02-3** and **Open Item 03.09.02-1,** respectively.

Structural fatigue and impact and fretting wear damage were estimated for components with nonnegligible TB-induced vibration amplitudes. High cycle fatigue caused by alternating stresses was expected to be insignificant because of the low coolant flow speeds in the NuScale reactor. However, some components (most notably the ICIGTs, CRD shafts, CRAGTs, and HCSG tubing) were expected to contact adjacent supports during normal operation. Peak amplitude was assumed to be 5 times the predicted root mean square amplitude, which captures a statistically appropriate number of peak occurrences and is reasonable because Au-Yang's workbook recommends 5 times the predicted root mean square amplitude. Contact and fretting stresses were estimated using well-established empirical methods. However, the estimated contact frequencies (termed "average crossing frequency") of many components are much less than the fundamental structural resonance frequencies. This is nonphysical and may lead to underestimates of wear. The staff issued **RAI 427-9408, Question 03.09.02-73** and **Question 03.09.02-74** (ADAMS Accession No. ML18107A802), asking NuScale to address these nonconservative average crossing frequencies. The staff is tracking **RAI 427-9408, Question 03.09.02-73** and **Question 03.09.02-74,** as **Open Item 03.09.02-3** and **Open Item 03.09.02-1,** respectively.

Although there may be some risk of structural wear caused by TB, FIV risks are much higher for stronger forcing mechanisms like VS and FEI. If the frequency of VS aligns with those of structural resonances and if the impedance of those resonances is small, lock-in can occur and cause significant vibration and damage. Structural impedance at resonance is related to the mass-damping parameter in ASME Code, Section III, Appendix N. In addition, if velocities are high enough to induce FEI in arrays of tubes (like the HCSG tubing), even higher vibrations and damage could occur. All components subjected to cross-flow were screened for susceptibility to VS. The only components without significant margin against VS/lock-in are the upper regions of the ICIGTs and CRD shafts and the lower HCSG tubing. The HCSG tubing was also assessed for susceptibility to FEI.

The upper portions of the ICIGTs and CRD shafts are subjected to primary coolant cross-flow and VS as the coolant flows radially outward from the riser region and transitions into downward flow through the outer annulus, which contains the HCSGs. In addition, the lower HCSG tubing is in cross-flow because the primary coolant flows downward and has no downstream structures to break up any vortices; therefore, it is subject to VS. All SG tubing may experience FEI at and above critical flow velocities. NuScale acknowledged that these components are at risk and reported the following small margins against these FIV mechanisms in accordance with ASME B&PV Code, Appendix N, VS/lock-in criteria:

- lower HCSG tubing VS/lock-in: less than 20 percent;
- HCSG tubing FEI: less than 10 percent;
- CRD shaft VS/lock-in: less than 25 percent; and
- ICIGTs VS/lock-in: less than 25 percent.

The HCSG tubing meets the ASME B&PV Code, Appendix N, VS avoidance criteria when 1.5-percent damping is assumed (which is higher than the 1-percent damping specified in RG 1.20, as discussed above). In addition, NuScale's FEI avoidance criteria are based on critical velocity data that may not be conservative. Measurements by S.S. Chen show that critical reduced velocities for HCSG tubing may be as low as 1.5 (S.S. Chen, "Tube Vibration in a Half-Scale Sector Model of a Helical Steam Generator," *Journal of Sound and Vibration* 91(4):539–569, 1983), which is lower than the value of [ ] assumed by NuScale.

The staff is concerned that the nonconservatisms in NuScale's analysis methodologies and assumptions stated above reduce or eliminate the reported margins against VS and FEI. In addition, NuScale tested sections of the HCSG tubing in prototypic conditions in the SIET TF-2 test, and the staff observed peaks in the tubing vibration spectra, which may result from VS or FEI, or both. (NuScale recently submitted TR-0918-60894, which includes updated assessments of these peaks; the staff is currently reviewing this report.) No such testing has been performed on the ICIGTs and CRD shafts; however, limited factory tests are planned for these components (see the section "Benchmark and Testing" below). The staff issued **RAI 427-9408, Question 03.09.02-73** and **Question 03.09.02-74** (ADAMS Accession No. ML18107A802), asking NuScale to address these specific component risks. The staff is tracking **RAI 427-9408, Question 03.09.02-73** and **Question 03.09.02-74,** as **Open Item 03.09.02-3** and **Open Item 03.09.02-1,** respectively.

VS/lock-in and FEI can lead to significant vibration amplitudes up to the diameter of the driven structure. However, the ICIGTs, CRD shafts, and HCSG tubing have small clearances at their respective supports. If VS or FEI, or both, occur, the adjacent supports will act as "snubbers" to restrict motion and prevent full lock-in from occurring. However, these repeated impacts could induce damage and failure quickly (in months or even weeks). The applicant has not submitted a forced response or impact and damage analyses for VS. Because the margin against VS/lock-in is less than 25 percent, forced response is nonnegligible and will be higher than that induced by TB. The staff issued **RAI 427-9408, Question 03.09.02-73** and **Question 03.09.02-74** (ADAMS Accession No. ML18107A802), asking the applicant to submit forced response VS analyses for the ICIGTs, CRD shafts, and lower HCSG tubing. In addition, the staff requested high cycle fatigue assessments that include stress amplifications at welds and joints and impact and fretting wear. The staff is tracking **RAI 427-9408, Question 03.09.02-73** and **Question 03.09.02-74,** as **Open Item 03.09.02-3** and **Open Item 03.09.02-1,** respectively.

AR issues in nuclear power plants are usually associated with flow instabilities that form over side openings in 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. The flow instabilities occur when a half or full wavelength of the vortices shed from the leading edge of a side branch coincide with the diameter of the opening. The half-wavelength instability is strongest (the fundamental) and is most likely to lock-in to any acoustic modes within the side branch. However, secondary (full wavelength) instabilities have also been observed in nuclear power plants and can be strong. The following components were evaluated for ARs:

- containment system and HCSG system steam piping, including nozzles and the MS isolation valves;
- HCSG steam plenum;
- DHRS steam pipe from the containment tee to DHRS actuation valves;
- DHRS condensate piping from the HCSG system feedwater piping tee to the DHRS condenser;
- RRV ports; and
- instrument ports.

The only locations in the NuScale piping that may be susceptible to ARs are in the DHRS lines: (1) a side branch near the DHRS actuation valve and (2) the condensate lines near the condensers. NuScale evaluated only the susceptibility to the primary flow instability and estimated a 20-percent margin against ARs for the side branch to the DHRS actuation valve. However, the staff is concerned that the secondary instability might also lock-in for the DHRS condensate piping from the HCSG system feedwater piping tee to the DHRS condenser. The staff issued **RAI 386-9316, Question 03.09.02-54** (ADAMS Accession No. ML18072A149), asking NuScale to assess this possibility. The staff is tracking **RAI 386-9316, Question 03.09.02-54,** as **Open Item 03.09.02-4** to address this concern.

NuScale did not provide analytical evaluations of LFI in the CVAP. However, NuScale stated that most components with leakage flowpaths have very low pressure differentials to ensure that leakage flow rates are very small; therefore, LFI is unlikely. However, NuScale did not submit quantitative information that demonstrates that RVIs should not be susceptible to LFI. The one exception is the array of SGIFRs that diffuses inlet flow and mitigates the possibility of density wave oscillation within the HCSG tubing. Various flow restrictor designs were not analyzed, but they were tested. NuScale submitted TR-0918-60894, which includes the SGIFR design concept LFI separate effects test results. The staff examined these test results during the 2018 audit; the audit report (ADAMS Accession No. ML18333A221) describes the staff's evaluation in detail. The final SGIFR design is based on a concept that showed no signs of LFI or any other significant FIV. The staff issued a follow-up **RAI 386-9316, Question 03.09.02-55** (ADAMS Accession No. ML18072A149), asking NuScale to provide the test plan for the final design tests. Finally, the staff issued **RAI 427-9408, Question 03.09.02-76** (ADAMS Accession No. ML18107A802), asking NuScale to provide details on its LFI screening of other RVIs. The staff is tracking **RAI 386-9316, Question 03.09.02-55,** and **RAI 427-9408, Question 03.09.02-76,** as **Open Item 03.09.02-5** and **Open Item 03.09.02-6,** respectively.

<u>Benchmarking and Testing</u>

NuScale has performed limited testing to benchmark its FIV analysis methodologies and has relied more heavily on screening and analysis results to identify RVI, piping, and HCSG components that are at risk of damage resulting from FIV and to identify the analysis areas that require validation testing. The current testing (as described in TR-0918-60894) is as follows:

- benchmark testing of the following:
    - HCSG tubing (SIET TF-1, TF-2, and TF-3 buildout for modes and damping)
    - SGIFR design concepts (for LFI)
- validation testing of the following:
    - HCSG tubing with prototypic supports (SIET TF-3 for modes, damping, VS, and FEI)
    - final SGIFR design (for LFI)
    - DHRS piping during initial plant startup (for ARs)
- planned additional factory testing, as described in the CVAP, for the following:
    - CRD shaft in-air resonance frequencies
    - ICIGT in-air resonance frequencies

Some of the benchmark testing has been completed (except for complete TF-3 modal and flow testing); TR-0918-60894 describes the results of the benchmark tests. As mentioned previously in this SER, several candidate SGIFR designs were tested for LFI and TB; the chosen design did not experience any significant vibration. Unexpected peaks appeared in HCSG tubing vibration spectra that may have been caused by VS/lock-in or FEI, or both (TF-2 testing), along with unexpected strong forces induced by two-phase secondary flow within the tubing (TF-1 testing). The applicant has not yet addressed these unexpected FIV effects in the NuScale design. Limited factory tests of the in-air modal behavior of the CRD shaft and ICIGTs are planned, but they will not be conducted until after the DC review. NuScale plans to submit TR-0918-60894, Revision 1, to address the planned factory test conditions, instrumentation, expected behavior, and acceptance criteria. The staff will assess the accuracy of the ICIGT and CRD shaft FIV analyses upon receipt of TR- 0716-50439, Revision 2.

The staff's evaluation of NuScale's FIV analyses and limited assessments of SIET TF-2 separate effects testing indicates that the HCSG tubing is at risk of VS/lock-in and FEI. Therefore, the upcoming SIET TF-3 modal validation tests are critical to confirm the adequacy of the HCSG design. These modal tests will be performed at different stages of TF-3 construction; test results will be compared to corresponding FE models. Current FE models assume pinned boundary conditions of the tubing at the supports, which leads to low resonance frequencies that may be overly conservative (the lower frequencies are closer to the flow VS and FEI frequencies). If testing establishes that the boundary conditions are stiffer (closer to clamped), the tubing resonance frequency estimates will increase, which will subsequently increase the margin against VS and FEI. NuScale has also assumed a 1.5-percent damping of HCSG tubing, which exceeds the 1-percent damping cited in RG 1.20. Modal testing will also be used to assess the actual damping. The testing will include variable preloading of the SG tubing (i.e., it will be pressed against the supports to emulate the effects of thermal expansion during plant operation). NuScale has not yet defined the actual preloading nor substantiated it with thermal expansion calculations. A future revision of TR-0918-60894 will provide the preloading definitions and justification. The staff's evaluation of NuScale's FIV analyses indicates that the CRD shaft and ICIGTs are also at risk of VS/lock-in. Therefore, factory testing and updated analyses are critical to confirm the long-term integrity of the CRDS and ICIGT designs. The staff has issued **RAI 427-9408, Question 03.09.02-73** (ADAMS Accession

No. ML18107A802), asking the applicant to submit the plans for the proposed factory testing. The staff is tracking **RAI 427-9408, Question 03.09.02-73,** as **Open Item 03.09.02-3**.

3.9.2.4.4 Flow-Induced Vibration Testing of Reactor Vessel Internals

DCA Part 2, Tier 2, Section 3.9.2.4, and the CVAP report state that the CVAP vibration measurement program consists of separate effects testing, factory testing, and initial startup testing. The CVAP report states that NuScale will submit two additional testing reports to the NRC. The first report contains the planned measurement program details for each prototype test and has been submitted (TR-0918-60894). The second report will include the posttest evaluation of the testing completed to support the measurement program and will be submitted after completion of the first validation testing and subsequently updated after completion of each validation test, as well as after the initial startup testing.

TR-0918-60894 describes an SIET TF-3 test for the HCSG assembly, which will be conducted before the initial startup testing. Performing this "validation testing" in a separate facility will allow more rigorous measurements with extended instrumentation and over much higher flow rate conditions than those possible in the actual plant. The expanded instrumentation and higher flow rates will allow NuScale to more strongly establish the ranges of operating conditions under which VS and FEI will not occur. However, the TF-3 flow validation testing results are currently not expected to be available during the DC review. This delay in the availability of the results will affect the staff's assessment of the HCSG FIV. The staff's 2018 CVAP audit report includes guidance on specific information needed to provide reasonable assurance that the HCSG design is not susceptible to VS or FEI, without the need for the TF-3 test, which then could be primarily used for design validation. The staff will evaluate future revisions of TR-0918-60894 to ensure that the flow test conditions, instrumentation, and acceptance criteria are appropriate. NuScale plans to provide this information in its response to **RAI 427-9408, Question 03.09.02-74,** and in the revision to TR-0918-60894. The staff is tracking this issue as **Open Item 03.09.02-1**.

Validation testing of the final SGIFR design will be performed before initial startup testing to obtain early validation of the SGIFR design, obtain more data than would be possible in the prototype initial startup testing, and evaluate more operating conditions. A facility like the one used to screen early SGIFR design concepts will be used but with expanded vibration and flow testing, as described in TR-0918-60894. In particular, the fundamental bending modes of the SGIFR will be measured along with FIV over a wide range of flow rates that significantly exceed those of normal plant operation. Testing will be at prototypic temperatures lower than prototypic pressures, which is acceptable for LFI assessments. Tests will be conducted for varying through-bolt compressions and radial eccentricities, including cases in which the SGIFR contacts the SG tube wall. Each test duration will ensure at least 1×10^6 cycles of vibration. This testing will establish that the SGIFR is not susceptible to LFI at normal plant operating conditions. In addition, because the CVAP FIV assessments included the SGIFRs, they are part of the NuScale inspection plan in accordance with TR-0918-60894, Table 7-1.

Planned initial startup testing of the prototype NPM for FIV mechanisms of RVIs is limited to (1) performing a flow test of the DHRS and steam piping to confirm the lack of significant ARs and (2) identifying and localizing any unexpectedly strong FIV effects. Instrumentation has not yet been specified for RVI components, including the CRD shafts, ICIGTs, CRAGTs, HCSG inlet flow restrictors, or the HCSG tubes—all of which are at risk for FIV-induced damage. In addition, instrumentation has not yet been specified for the upper or lower riser, which are separated by a slip joint that has not been evaluated for LFI. TR-0918-60894, Revision 0, states that the RVI instrumentation and allowable limits will be specified in a future revision. TR-0918-60894 provides initial specifications for testing DHRS piping to assess any significant AR effects. Because any strong AR will lead to high vibrations and internal pressures, both are monitored. Several accelerometers will be mounted to the DHRS piping to monitor vibration. Either pressure taps (which will penetrate the piping to directly measure pressures) or circumferential arrays of strain gauges (which indirectly measure pressure through hoop strain) will be installed. The staff finds that either measurement procedure is acceptable because the pressure taps measure the pressures directly, and strain gauge arrays have been used successfully in many boiling-water reactor MS measurements during recent extended power uprates. This combination of instrumentation is sufficient to determine whether significant ARs are present.

The testing duration is determined based on the lowest structural natural frequency from the analyses, and a goal of 1 million cycles of vibration is to be achieved. The total test time will be less than 2.5 days. DHRS AR testing will include varying flow rates to identify both second-order and first-order shear layer instabilities that might excite ARs in standpipes. Although flow rates that lead to second-order ARs should be identifiable (because they occur at flow velocities lower than the maximum design speed), it is expected that first-order shear layer ARs will not occur because those corresponding flow speeds are expected to be slightly above the maximum design speed.

Initial startup testing and acceptance limits have not yet been fully described for RVIs. Allowable limits for AR testing of the DHRS for both first- and second-order ARs have also not yet been defined. Therefore, the staff has issued **RAI 427-9408, Question 03.09.02-77** (ADAMS Accession No. ML18107A802), asking NuScale to provide details and pretest predictions for the initial startup testing scope that will give reasonable assurance that RVIs, DHRS piping, and the HCSG will not experience significant FIV-induced damage and failure. The staff is tracking **RAI 427-9408, Question 03.09.02-77,** as **Open Item 03.09.02-7**.

In accordance with TR-0918-60894, Section 7, "Inspection Program," NPM components that were evaluated for FIV will be inspected after initial startup testing, following the guidelines and requirements provided in ASME B&PV Code, Section III, Paragraph NG-5111, "General Requirements," and Paragraph NB-5111, "Methods," and using the methods defined in ASME B&PV Code, Section V, "Nondestructive Examination." VT-1 and VT-3 were used to perform the visual inspections, as defined by ASME B&PV Code, Section XI, Subarticle IWB-2500, "Examination and Pressure Test Requirements," Table IWB-2500-1 (B-N-1, B-N-2, B-N-3), "Examination Categories B-N-1, Interior of Reactor Vessel; B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels; B-N-3, Removable Core Support Structures." The inspected areas included 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 are 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. NuScale also plans periodic inservice inspections of the RVIs. The staff finds that the inspection methods and areas are consistent with those in the previous applications and with the guidance in RG 1.20.

DCA Part 2, Tier 2, Revision 1, Section 3.9.2.4, COL Item 3.9-1, states that a COL applicant that references the NuScale Power Plant DC will provide the applicable test procedures before the start of testing and will submit the test and inspection results from the CVAP for the NPM in accordance with RG 1.20. The staff finds COL Item 3.9-1 acceptable because the COL item description is consistent with the provision in RG 1.20 on submitting the CVAP.

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

This section contains three subsections. The first subsection contains the staff's evaluation of the seismic analysis and the short-term transient analysis of the NPM. The second subsection contains the staff's evaluation of the RVI components stress analysis under the Service Level D Faulted Condition. The third subsection contains the staff's evaluation of RPV primary stress under the Service Level D Faulted Condition.

<u>Seismic and Short-Term Transient Analysis of NuScale Power Module</u>

TR-0916-51502 documents the NPM seismic analysis. The report contains analysis methodology, input motion, structural modeling of the major NPM components (i.e., the containment, reactor vessel, upper RVIs, lower RVIs, and CRDM) and analysis results, including displacements, ISRS, forces, and moments at component interfaces. The major NPM components were modeled by ANSYS FE meshes. The calculated component interface forces, moments, and ISRS were used as input loads for the component-level stress analysis. This subsection evaluates the analysis methodology, structural modeling of the NPM component, and analysis results.

<u>Analysis Methodology</u>

The NPM seismic analysis methodology consists of the following steps:

(1) Create a 3-D NPM ANSYS model. The model consists of five submodels (i.e., containment with the surrounding pool water, reactor vessel, lower RVIs, upper RVIs, and CRDM). The five submodels are connected by coupling nodes, constraint equations, contact elements, and fluid coupling to form the 3-D NPM model.

(2) Create a simplified NPM ANSYS beam model that is dynamically equivalent (dry and wet submerged conditions) to the 3-D NPM model. Modal, harmonic, and transient analyses are performed to tune the beam model to match the dynamic response of the 3-D NPM model.

(3) Convert the simplified NPM ANSYS beam model to the simplified NPM SAP2000 beam model for the system for analysis of soil-structure interaction (SASSI) SSI analysis.

(4) Perform an SSI analysis with a SASSI model, including an NPM SAP2000 beam model in each of the 12 reactor bays along with the SAP2000 RXB model. The results of the SSI analysis indicate that NPM 1 and NPM 6 have the highest seismic response among the 12 NPMs.

(5) Create an ANSYS NPM entire pool model. In the NPM entire pool model, the 3-D NPM model is modified to include the entire volume of the reactor pool. The NPM entire pool model only includes NPM 1 or NPM 6.

(6) Perform a time history analysis of the NPM entire pool model by applying the calculated acceleration time histories from the SSI analysis as boundary conditions at the pool floor and walls, as well as at the NPM supports. For the 11 NPMs not included in the NPM

entire pool model, the calculated NPM centerline acceleration time histories from the SSI analysis are applied to the surface of the fluid surrounding the NPM to simulate the effects of the missing NPMs. The NPM entire pool model is analyzed for six cases with the following conditions:

- one certified seismic design response spectra (CSDRS) compatible 0.5g horizontal ground motion and one CSDRS compatible 0.4g vertical ground motion;
- one soil type;
- two NPM module locations (NPM 1 and NPM 6);
- two RXB concrete conditions (cracked and uncracked); and
- one nominal NPM stiffness for the uncracked concrete condition and two NPM stiffnesses for the cracked concrete condition (nominal NPM stiffness and NPM stiffness adjustment = 1/1.3 = 77 percent of NPM nominal stiffness).

(7) Generate seismic loads within the NPM from the six analysis cases.

(8) Perform a component-level stress analysis using the maximum seismic loads in step 7.

The staff considers the methodology of the NPM seismic analysis acceptable because it is consistent with common engineering practice that an SSI analysis is performed first, and the output motions from the SSI analysis are then used as input motions for the reactor systems analysis. However, the staff is concerned that the scope of the NPM seismic analysis is inadequate because it only considers the NPM in operation while immersed in the pool water. Because one of the 12 NPMs is being refueled during normal operation, the staff issued **RAI 202-8911, Question 03.09.02-19** (ADAMS Accession No. ML17237C062), asking the applicant to justify why it did not include in the NPM seismic analysis a scenario in which the NPM is in the refueling bay, in dry dock, or in transit. In its response to **RAI 202-8911, Question 03.09.02-19** (ADAMS Accession No. ML17297B951 (proprietary) and ML17297B950 (nonproprietary)), the applicant stated that the results of the NPM seismic analysis bound the results of the NPM placed in the refueling bay as well as suspended by the RXB crane in transition mode. The applicant presented the NPM seismic response in the operating bay and in the refueling bay in its response to **RAI 19-8769, Question 04.02-07** (ADAMS Accession No. ML17180A483). The staff verified that the results of the NPM seismic response in the operating bay bounds the results of the NPM in the refueling bay. The NPM seismic analysis in dry dock was not performed because the NPM is classified as a seismic Category III structure while it is in dry dock. The staff agrees that there is no need to perform an NPM seismic analysis while the NPM is in dry dock because the NPM contains no fuel while it is in dry dock. The applicant stated that, during the NPM transport mode, the NPM is isolated from the transmission of horizontal seismic loads because the module lifting adapter does not provide lateral restraint when the NPM is suspended by the building crane. The staff closed **RAI 202-8911, Question 03.09.02-19,** and issued **RAI 410-9310, Question 03.09.02-59** (ADAMS Accession No. ML18099A356), asking the applicant to address the NPM seismic response under vertical seismic loading when suspended by the RXB crane. In its response to **RAI 410-9310, Question 03.09.02-59** (ADAMS Accession No. ML18159A356), the applicant stated that vertical seismic response of the NPM when it is suspended by the building crane is not evaluated because the NPM is suspended by the building crane only for short time durations.

The applicant further stated that the U.S. operating nuclear fleet does not perform a seismic evaluation of the reactor components and nuclear fuel during refueling operations based on their short time durations in the refueling phases. The applicant stated that, for a 12-module NPM power plant, the time in refueling transition configurations is less than 5 days per year, based on the refueling of six modules per year and an estimated 20 hours of crane movement per NPM refueling, which is much less than the average refueling duration of 40.2 days per year of a U.S. PWR. The applicant incorporated the response in TR-0916-51502, Revision 1. The staff found the applicant's response to be acceptable because of the short time durations that NPMs are suspended by the building crane; therefore, **RAI 410-9310, Question 03.09.02-59,** is resolved and closed. SER Chapter 9 addresses the review of the RXB crane.

<u>NuScale Power Module Modeling</u>

In the structural modeling of the NPM, a full 360-degree 3-D NPM ANSYS model was created for the NPM seismic analysis. The staff finds that this is appropriate because the NPM does not have symmetric conditions for a half model or a quarter model. The 3-D NPM ANSYS model consists of five submodels representing the CNV and surrounding pool water in a single bay, RPV, lower RVI, upper RVI, and CRDM. In each submodel, the ANSYS eight-node shell element and eight-node solid element are used for modeling the geometry. A 360-degree shell cross section is generally represented by 16 shell elements in the circumferential direction. The mass of the shell and solid element was adjusted to account for the mass of boltings, piping fluid, and cables that the submodels excluded. The pool water is modeled by eight-node acoustic fluid elements (FLUID30). The five submodels are connected by coupling nodes, constraint equations, contact elements, and fluid coupling to form the 3-D NPM model.

The CNV submodel consists of the CNV and the top auxiliary mechanical access structure. The RPV submodel consists of the RPV shell, top head, upper and lower supports, feedwater and MS plenums, and CRDM support structure. At the bottom of the RPV, the nodes associated with the RPV alignment feature and the nodes associated with the CNV alignment feature were coupled in the horizontal direction, not in the vertical direction to allow thermal expansion. The RPV upper support was coupled to the CNV ledges in the vertical and circumferential directions, not in the radial direction to allow thermal expansion.

The lower RVI submodel consists of the lower riser shell, the core support assembly (i.e., core barrel, lower and upper core plates, and reflector), fuel assemblies, and CRD shaft supports. Seismic inertia loads of the lower RVI were transferred to the RPV through several connections. In the horizontal direction, the upper support blocks of the lower RVI were coupled to the RPV by no-separation contact elements between the four pairs of mating surfaces. The lower core plate tabs of the lower RVI were coupled to the core support block assembly of the RPV. A simplified beam model represented the 37 fuel assemblies with mass and stiffness tuned to match the fuel assembly frequencies provided by the fuel vendor. The simplified fuel model was connected to the upper and lower core plates of the lower RVI. The reflector of the lower RVI was connected to the core barrel by no-separation contact elements.

The upper RVI submodel consists of the upper riser shell and five CRDS supports. Half of the SG mass was applied to the upper RVI, and the other half was applied to the RPV. Radial coupling was provided between the upper riser and the RPV for transferring the SG seismic loads. Bellows were installed between the upper riser and the lower riser. The bellows allow for vertical thermal growth while limiting relative horizontal displacements between the upper riser and the lower riser. The upper riser was coupled to the lower riser in the horizontal direction only. The fluid in the annular region between the risers and the RPV was accounted for by fluid coupling between the risers and the RPV. Fluid coupling nodes on the inner and outer surfaces of the annulus were specified at 23 elevations along the annulus.

The CRDM submodel consists of 16 CRDM assemblies that were modeled by pipe, beam, and mass elements. The tops of the CRDMs were coupled laterally to the CNV top head, and the mid-heights of the CRDMs were coupled laterally to the CRDM support frame on top of the RPV. The staff's concerns on structural modeling of the NPM components are described below.

TR-0916-51502, Revision 0 issued January 2017 (ADAMS Accession Nos. ML17010A434 (proprietary) and ML17010A433 (nonproprietary)), Section 4.1.3.1, "Lower Reactor Vessel Internals Geometry, Mesh and Mass," states that the lower RVI submodel geometry was based on the lower riser and core support drawings. The computer-aided design model used to generate the drawings was defeatured and simplified to reduce the element count of the mesh. The staff's concern is whether the provision in SRP Section 3.9.2, Revision 3, can be met with the reduced element count in the NPM mesh. SRP Section 3.9.2, Revision 3, states that the number of elements is adequate when additional degrees of freedom do not result in more than a 10-percent increase in responses. The applicant supplemented its DCA by letter dated November 20, 2017 (ADAMS Accession No. ML17297B951 (proprietary) and ML17297B50 (nonproprietary)), in which it stated that a mesh sensitivity study of the NPM model has been performed. The modal analysis was carried out for a refined NPM model with a tenfold increase in the number of elements. A comparison of the modal response (frequency and associated mass participation) with the original coarse NPM model revealed that the difference was within 10 percent. The applicant provided a table that lists the calculated dominant frequencies of major NPM components of the original coarse NPM model and the refined NPM model. The staff verified the differences of the major dominant frequencies are all within 10 percent and, therefore, finds that the applicant met the provision in SRP Section 3.9.2, Revision 3. In the same letter, the applicant stated that the following components were modeled within the lower RVI submodel and provided the associated element types:

- lower riser and core barrel (shell element); and
- upper core plate; upper CRD shaft support; CRAGT support plate; lower core plate; reflector blocks (upper, intermediate, and lower); and upper support blocks (solid element).

The staff noted that the lower RVI submodel includes the major components in the lower RVI. However, the modeling does not consider the 0.125-inch fluid gap between the core barrel and reflector. During the 2017 audit (ADAMS Accession No. ML18023A091), the staff noted that the narrow fluid gap between the core barrel and reflector can affect the natural frequency of the core barrel and reflector significantly. Without considering the fluid gap, the core barrel has a fundamental frequency above [ ] hertz, and the reflector has a much higher fundamental frequency than that (the data are from EC-A023-3535, "RVI Turbulent Buffeting Degradation Evaluation," Revision 0, dated November 18, 2016). With the fluid gap considered in the modeling, the frequencies of the first five modes of the core barrel-fluid, gap-reflector coupled system are much lower (between [ ] hertz (the data are from ER-A010-2157, "Methodology Development for Hydrodynamic Effect Evaluation for Reflector and Core Barrel," Revision 0, dated November 30, 2015). The staff issued **RAI 410-9310, Question 03.09.02-70** (ADAMS Accession No. ML18099A356), asking the applicant to justify that the omission of the fluid gap between the core barrel and the reflector in the lower RVI submodel provides conservative NPM seismic analysis results. In its response to **RAI 410-9310, Question 03.09.02-70** (ADAMS Accession No. ML18278A247), the applicant stated that, to address the differences between the modeling approach with and without a fluid gap, it updated the lower RVI submodel to capture the effect of the fluid gap using the Fourier nodes methodology. Resultant forces at the top and bottom of the core barrel were found to be less conservative when using the no-separation contact element between the core barrel and the reflectors (i.e., the fluid gap was not considered). The applicant further stated that, instead of using the no-separation contact, the fluid gap was modeled using the Fourier node method and indicated that TR-0916-51502, Revision 2, will describe the updated lower RVI submodel. The staff finds the use of the Fourier node method in modeling the fluid gap between the core barrel and reflectors acceptable because the Fourier node method can be used to simulate the dynamic response of two concentric cylinders with fluid gap. The staff is tracking **RAI 410-9310, Question 03.09.02-70,** as **Confirmatory Item 03.09.02-1**, pending the applicant's incorporation of the updated lower RVI submodel in TR-0916-51502, Revision 2.

TR-0916-51502, Revision 0, Section 4.1.3.3, "Lower Reactor Vessel Internals Boundary Conditions," states that spring elements were added between the lower core plate tabs and the lower core support blocks of the RPV to represent the Belleville washers at these locations. The staff issued **RAI 410-9310, Question 03.09.02-62** (ADAMS Accession No. ML18099A356), asking the applicant to provide a spring constant of the Belleville washers. In its response and supplemental response to **RAI 410-9310, Question 03.09.02-62** (ADAMS Accession Nos. ML18207A526 and ML18305B315), the applicant stated that the Belleville washers were removed from the NuScale design and that the RPV lower core support blocks were replaced by the core support block assemblies with socket head cap screws and shear pins that provide vertical and circumferential restraints for the lower core support plate. Each core support block assembly consists of a top plate that sits on top of two gussets that are curved to match the profile of the bottom head. The applicant further stated that the Belleville washers were removed from the design because it identified an artificially strong AR that significantly increases vibrations and loads in the RVI. The assumption of the complete reflection of acoustic waves at the reactor pool floor caused the artificially strong amplitude of the resonance, although, in reality, the energy is partially reflected and partially absorbed by the concrete and surrounding structures and soil. After performing a detailed fluid-structure interaction (FSI) analysis, the artificial AR was attenuated, and the Belleville washers were no longer necessary. The artificial AR was attenuated by applying an acoustic-absorbing coefficient of 0.75 (TR-0916-51502, Revision 1) at the FSI interface of the RXB floor. On December 19, 2018, the staff audited the detailed FSI analysis that yields the acoustic-absorbing coefficient. The staff evaluated NuScale's seismic analysis methods, models, and results used to develop the acoustic attenuation coefficient (see the audit report at ADAMS Accession No. ML19072A136). NuScale modified the acoustic coefficient from 0.75 to 0.40 during the audit. The absorption coefficient of 0.40 applied to the bottom of the pool model is conservative and reasonable based on the relative acoustic impedances of water and concrete and on NuScale's analyses of a detailed model of the overall building. The vertical AR in the pool is appropriately damped by the applied absorption coefficient, which reduces the calculated NPM vibration under seismic loading. As the applicant removed the description of the Belleville washers from DCA Part 2, Tier 2, Revision 2, the staff considers **RAI 410-9310, Question 03.09.02-62,** to be resolved and closed.

TR-0916-51502, Revision 0, Section 4.1.4.1, "Upper Reactor Vessel Internals Geometry, Mesh and Mass," states that the upper RVI submodel geometry was based on the upper riser drawings. The computer-aided design model used to generate the drawings was defeatured and simplified to reduce the element count of the mesh. The staff is concerned that oversimplification in the upper RVI submodel can affect the results of the NPM seismic analysis. In a letter dated October 24, 2017 (ADAMS Accession No. ML17297B950), the applicant stated that the upper RVI submodel mainly consists of the upper riser shell and five CRDS supports. The bottom of the upper riser shell is attached to a cone that connects the upper riser to the lower riser. There are bellows between the upper riser shell and the upper riser conic section. The bellows allow for vertical thermal growth while limiting relative horizontal displacements between the upper riser and the lower riser. The applicant further stated that the geometry of the bellows was not modeled; its effect was captured by coupling the upper riser and lower riser in the horizontal direction. The vertical direction is not coupled for thermal growth. The bellows have a complex geometry. In letters dated July 19, 2018, and October 15, 2018 (ADAMS Accession Nos. ML18201A268 and ML18288A270, respectively), the applicant provided a sketch of the bellows assembly, which consists of the overlapping upper and lower lateral restraints (i.e., sliding surfaces) and the bellows vertical expansion structure (i.e., convolutions). The applicant stated that the lateral restraints provide structural support for the CRD shafts and are classified as a seismic Category I component. The applicant has stated that it intends to include the stress results of the lateral restraints in its response to **RAI 202-8911, Question 03.09.02-18**. The staff will evaluate the stress results later in this section of the SER. The bellows convolutions prevent bypass flow between the cold and hot legs with no structural support function during normal operation and are classified as seismic Category II components. The applicant also stated that the failure of bellows convolutions will not affect natural circulation of the reactor because of the narrow fluid path of the lateral restrains and low differential pressure in the primary loop. The differential pressure between the hot leg and cold leg at the bellows is small (less than [(   ] Kilopascal (KPa))). These conditions result in an insignificant amount of bypass flow from failure of the nonseismic, convoluted portion of the bellows. The staff agrees that the bellows convolutions can be classified as seismic Category II components because of the narrow fluid path of the lateral restrains and low differential pressure and because they have no structural support function.

TR-0916-51502, Revision 0, Section 4.1.8.2, “Upper Reactor Vessel Internals Boundary Conditions,” states that half of the SG horizontal mass is applied to the riser and that the other half is applied to the inner surfaces of the RPV. The vertical mass is applied to the upper SG supports only to represent the floating vertical connection at the bottom cantilever interface. The staff finds the description insufficient to reach a safety finding. The staff noted that the SGs are large components and that the 3-D NPM model did not include them. The staff also noted that the applicant ignored the SG stiffness without assessing its effect on the results presented in TR-0916-51502. In a letter dated June 20, 2018 (ADAMS Accession No. ML18171A409), the applicant provided a calculation of the bending stiffness of the RPV and the aggregated bending stiffness of 84 SG tube support columns. The calculation showed that the bending stiffness of the tube support columns is 4 orders of magnitude lower than the RPV. The applicant concluded that neglecting the SG horizontal stiffness in the NPM model is justified. For the vertical SG stiffness, the staff verified the following criteria in SRP Section 3.7.2, “Seismic System Analysis,” for decoupling a subsystem from the primary system:

i. If $R_m < 0.01$, decoupling can be done for any R_f.

ii. If $0.01 < R_m < 0.1$, decoupling can be done if $0.8 > R_f > 1.25$.

iii. If $R_m > 0.1$, a subsystem model should be included in the primary system model.

Where:

R_m = the ratio of the total mass of the supported subsystem to the total mass of the supporting system.

R_f = the ratio of the fundamental frequency of the supported subsystem to the dominant frequency of the supporting system.

The staff calculated the SG vertical frequency, R_m, and R_f based on the mass of the SGs and the aggregated vertical stiffness of the 168 SG tube support columns. The R_m and R_f values meet the decoupling criteria in (ii). The staff concludes that the SG vertical stiffness can also be excluded in the NPM model. Therefore, the staff finds that the SGs can be excluded in the 3-D NPM model.

Simplified NuScale Power Module Model

TR-0916-51502 states that two simplified NPM beam models were developed. The ANSYS model was created first and tuned to match the dynamics characteristics of the 3-D NPM model under dry and wet submerged conditions. The tuned ANSYS model was then converted to the simplified SAP2000 NPM model for use in the SSI analysis along with the SAP2000 RXB structure model. The simplified NPM models consist of the beam, spring, and mass elements. The CNV and RPV are represented by distinct beams that are coupled together using stiff spring elements. Torsional mass moment of inertia was assigned in the mass elements to account for torsional effects. A set of tuned point masses and springs represents the dynamic characteristics of upper and lower RVIs. The dynamic characteristics of the simplified models were tuned to match the 3-D NPM model under dry and wet submerged conditions. Tuning was accomplished by changing the elastic modulus material properties and by adding spring-mass systems to capture missing modes. After repeated tuning, the dominant natural frequencies and associated modal mass of the simplified models were mostly within a 10-percent difference compared to the 3-D NPM model in three directions. Static, harmonic, and transient analyses were also performed. The results of the two simplified models agree well with the results of the 3-D NPM model. Based on the comparison, the staff determined that the applicant properly created the simplified SAP2000 NPM model for use in the SSI analysis. The SSI model contains 12 SAP2000 NPM models, one in each reactor bay. The results of the SSI analysis indicated that NPM 1 and NPM 6 have the highest seismic response among the 12 NPMs.

NuScale Power Module Entire Pool Model and NuScale Power Module Seismic Loads

Based on the results of the SSI analysis, the forces at the CNV support skirt and lug supports of the NPM at operating bays 1 and 6 bound those of the other NPM locations. NPMs at operating bays 1 and 6 were selected for the NPM seismic analysis to bound the seismic response of all NPMs. Two ANSYS NPM entire pool models were created for generating NPM seismic loads for component-level stress analysis. Model 1 has an NPM at operating bay 1, and Model 2 has an NPM at operating bay 6. In both models, the 3-D NPM model was modified to include the entire volume of the reactor pool. A contact element was used between the RXB floor and the CNV skirt to capture any uplift of the NPM. A nonlinear time history analysis was performed by applying the calculated acceleration time histories from the SSI analysis as boundary conditions at the pool floor and walls, and at the NPM supports. For the 11 NPMs not included in the entire pool model, the calculated NPM centerline acceleration time histories from the SSI analysis were applied to the surface of the fluid surrounding the NPM to simulate the effects of missing NPMs. For each NPM entire pool model, the following three conditions were considered:

(1) RXB uncracked concrete condition with no adjustment of NPM stiffness;

(2) RXB cracked concrete condition with no adjustment of NPM stiffness; and

(3) RXB cracked concrete condition with adjustment of NPM stiffness (NPM stiffness adjustment = 1/1.3 = 77 percent of the NPM nominal stiffness).

A total of six cases were performed for generating NPM seismic loads for the two NPM entire pool models.

Based on the results of the six analysis cases, the following seismic loads were generated at various locations within the NPM for component-level stress analysis:

- displacement and acceleration time histories and broadened ISRS at 33 locations within the NPM;
- maximum seismic forces and moments at 77 interfaces between the NPM components;
- maximum seismic forces and moments within 22 NPM component sections; and
- maximum seismic forces at four NPM support locations.

The maximum NPM uplift occurred in the case of the NPM in operating bay 6, with a cracked concrete condition and with no adjustment of NPM stiffness. The paragraphs below discuss the staff's concern with regard to the NPM entire pool seismic analysis, system damping, and method of generation of NPM seismic load.

TR-0916-51502, Revision 0, Section 7.2, "Response Spectrum Analysis Method," states that, from time history analyses of the 3-D NPM in the entire pool model, time histories at locations of equipment supports within the NPM were calculated. Response spectra were then enveloped and broadened in accordance with American Society of Civil Engineers (ASCE) 4-13, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," issued on July 4, 2013 to provide ISRS for use in the design of the SSCs supported within or directly on the NPM. DCA Part 2, Tier 2, Section 3.7.2.5, "Development of In-Structure Floor Response Spectra," states that development of ISRS follows the guidance in RG 1.122, Revision 1. The staff finds that the use of ASCE 4-13 for generating the NPM ISRS is inconsistent with the guidance in RG 1.122, which is used for generating the rest of the ISRS within the RXB, as stated in DCA Part 2, Tier 2, Section 3.7.2.5. The major difference between ASCE 4-13 and RG 1.122 is that ASCE 4-13 permits a 15-percent reduction in the narrow frequency peak amplitude if certain conditions are met. The 15-percent reduction of the narrow frequency peak amplitude is not consistent with the RG 1.122 criteria. The use of a 15-percent reduction of the narrow frequency peak amplitude may result in nonconservative seismic results.

However, the applicant subsequently addressed the staff's concerns in a letter dated October 24, 2017 (ADAMS Accession No. ML17297B950), to supplement the DCA, the applicant stated that there are six 3-D NPM seismic analysis runs. For each location direction and damping value, the response spectra were calculated for the six seismic analysis runs. An envelope spectrum was constructed by finding the maximum of the six response values at each spectral frequency point. The envelope spectrum was then broadened by ±15 percent to produce the design ISRS in TR-0916-51502, Revision 0. The applicant further stated that the design ISRS in TR-0916-51502, Revision 0, were generated using the guidance in RG 1.122, Revision 1. The reduction of narrow frequency peak amplitudes was not performed. The applicant provided a markup of TR-0916-51502, Revision 1, to include the information mentioned above. The staff found the ISRS development procedure acceptable because the design ISRS were constructed in accordance with the guidance in RG 1.122, with no reduction in the narrow frequency peak amplitudes. The generated seismic loads are conservative because the design ISRS is the envelope spectrum from all six analysis cases. The staff verified that the applicant updated TR-0916-51502 to state that the design ISRS in the NPM seismic analysis were generated using the guidance in RG 1.122, Revision 1. The applicant removed the description of ASCE 4-13 from the report.

TR-0916-51502, Revision 0, Section 8.0, "Three-Dimensional Seismic Model Analysis," states that the 3-D NPM in the entire pool model was analyzed using the CSDRS compatible Capitola time histories. Outputs of the analysis include the ISRS, time history data, relative displacements, and forces and moments within the NPM. The analysis did not consider the high-frequency certified seismic design response spectrum (CSDRS-HF) compatible time histories (i.e., the Lucerne time histories). The applicant included COL Item 3.9-12 in DCA Part 2, Tier 2, Section 3.9.1, "Special Topics for Mechanical Components," for the COL applicant to address CSDRS-HF input at hard rock sites. The staff finds COL Item 3.9-12 acceptable because it adequately addresses the NPM component design under CDSRDS-HF inputs.

TR-0916-51502, Section 8.4.2.3, "Forces and Moments at Component Interfaces," and Section 8.4.2.4, "Forces and Moments within Component Sections," state that the ANSYS FSUM command was used to select a set of nodes and elements for the nodal force and moment summations to encompass the entire cross section. All the elements and associated nodes on one side of the cross section were selected for the nodal force and moment summations by using the FSUM command. In a letter dated October 24, 2017 (ADAMS Accession No. ML17297B950), the applicant provided additional information on how the maximum forces and moments were determined. The applicant stated that nodal forces and moments associated with the elements adjacent to a cross section or interface were calculated for all time steps and that the maximum values were determined from all the time steps and all the analysis runs. The maximum forces and moments acting on the selected set of nodes from the selected elements on one side of the cross section cut were summed about a point at the centerline of the cross section or interface to obtain the resultant forces and moments. Only forces and moments acting on the selected nodes and elements contributed to the resultant forces. The staff finds this acceptable because the ANSYS FSUM command requires the user to define a selected set of nodes and elements for calculating resulting forces and moments acting on that set of nodes and elements. The applicant selected all the elements and nodes encompassing the entire cross section using the FSUM command to calculate the resulting forces and moments at component cross sections and interfaces.

TR-0916-51502, Revision 0, Section 7.4, "Uncertainties in the NuScale Power Module Subsystem Model," states that uncertainty in the input and assumptions used in the 3-D NPM model was accounted for by considering multiple analyses using ±30-percent variations of the stiffness properties of the model. TR-0916-51502, Revision 0, Section 8.4.3, "NuScale Power Module Seismic Analysis Results," states that maximum uplift displacements of the NPM were calculated for each of the six analysis runs. The maximum uplift occurred in the run representing the Capitola time history, soil type 7, with cracked RXB concrete on the NPM in operating bay 6 with a nominal NPM stiffness. TR-0916-51502, Revision 0, Section 8.0, further states that, for the cracked RXB concrete case, the NPM seismic analysis considered two NPM stiffnesses: (1) NPM stiffness adjustment = 1/1.3 = 77 percent of nominal NPM stiffness and (2) normal NPM stiffness with no adjustment made. Increasing the NPM stiffness may bring the NPM dominant frequency closer to the frequency of the cracked RXB. Therefore, in the calculation of the maximum NPM uplift, the staff believes that the NPM seismic analysis should also consider the case of 130 percent of the NPM stiffness for the cracked RXB concrete case. The results of 77-percent NPM stiffness could be nonconservative. The staff issued **RAI 202-8911, Question 03.09.02-43** (ADAMS Accession No. ML17237C062), asking the applicant to provide the maximum NPM uplift in the cracked RXB concrete case with an adjustment of +30-percent variation of the NPM stiffness (i.e., 130-percent NPM stiffness). The staff is tracking **RAI 202-8911, Question 03.09.02-43,** as **Open Item 03.09.02-8**.

TR-0916-51502, Revision 0, Appendix C, Table C-1, "Summary of NuScale Power Module Component Interfaces," states that the reflector blocks and lower core plate are stacked and restrained in the horizontal direction with alignment pins. The reflector blocks and lower core plate are not restrained in the vertical direction other than by gravity. On the interface between the upper riser and the lower riser, Table C-1 states that the lower riser conical section sits within the upper riser conical section. The upper riser is not restrained in the vertical direction other than by gravity and compression of the upper riser bellows, which keep the interface between the upper riser and lower riser closed. The staff is concerned that the reflector blocks and the upper riser could uplift in a seismic event. The staff issued **RAI 202-8911, Question 03.09.02-45** (ADAMS Accession No. ML17237C062), asking the applicant to provide the following information:

- TR-0916-51502, Revision 0, Figure B-18 (vertical ISRS at the top of the lower core plate), indicates that the vertical spectral acceleration at the top of the lower core plate at the high-frequency end (frequency = 100 hertz) of the ISRS is about 1.6g, which exceeds gravity acceleration. Discuss the possibility of an uplift occurring between the reflector blocks and between the reflector blocks and the lower core plate under the maximum vertical acceleration and its consequence.

- Provide the vertical ISRS and the maximum vertical acceleration at the interface between the upper riser and the lower riser. Discuss the possibility of an uplift of the upper riser from the lower riser under the maximum vertical acceleration and its consequence.

The staff is tracking **RAI 202-8911, Question 03.09.02-45,** as **Open Item 03.09.02-9**.

<u>System Damping</u>

TR-0916-51502, Section 8.1, "Transient Analysis," states that the NPM component seismic analysis uses 4-percent damping. RG 1.61, Revision 1, Table 6, "Damping Values for Mechanical and Electrical Components," recommends 3-percent SSE damping for pressure vessels and major pressure boundary components. The staff finds that using 4-percent damping in NPM subsystems and system analysis is reasonable and acceptable. The integrated NPM with many connections and internal structures is unlike traditional shell type pressure vessels, and use of higher damping 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.

<u>NuScale Power Module in the Reactor Flange Tool</u>

TR-0916-51502 documents the seismic analysis of the NPM in the reactor flange tool (RFT) for refueling. The NPM RFT model encompasses a subset of the NPM model, including the lower RPV, the lower RVIs, the core support structure with fuel, and a representation of the RFT. The RPV submodel was modified to incorporate the refueling ledge at the bottom of the RPV. The mass of the displaced pool water was accounted for by increasing the density of the RPV, RFT, and the lower RVIs. Acceleration time histories on top of the base mat at the RFT location were applied to the bottom of the RFT model. The analysis considered cracked and uncracked concrete in the RXB with normal and adjusted NPM stiffness. No uplift of the RPV from the RFT was assumed. Time history analysis was carried out with 4-percent system damping. The output consists of ISRS at the top of the lower core plate and at the bottom of the upper core plate. The applicant stated that it will introduce nonlinear contact elements at the interface of lower RPV and RFT to simulate uplift of the NPM and will document the results in TR-0916-51502, Revision 2. The staff will review TR-0916-51502, Revision 2, when it is available. The staff is tracking this as **Open Item 03.09.02-10**.

Short-Term Transient Analysis of the NuScale Power Module

TR-1016-51669 documents the NPM short-term transient analysis. MSPB, FWPB, and DBPB cause short-term transient events that result in an asymmetric cavity pressurization load between the CNV and RPV and blowdown load within in the RPV. DBPB can also cause the inadvertent opening of the RVV, RSV, and RRV and a CVCS pipe break. The report contains the analytical methods, benchmarking for validating the analysis methods, and the resulting asymmetric cavity pressurization load and blowdown load. These short-term transient structural loads were combined with the SSE load for evaluating stress on the NPM internals under the Service Level D Condition. The thermal-hydraulic code NRELAP5 and the ANSYS model calculate the short-term transient loads. NRELAP5 generates thermal-hydraulic boundary condition inputs for the ANSYS model, which calculates the short-term transient structural loads within the NPM, including forces, moments, and differential pressure loads.

Two ANSYS models were developed to determine the short-term transient structural loads. The first model was used for calculating the asymmetric cavity pressurization load between the CNV and the RPV. The second model was used for calculating the blowdown load within the RPV. There are three RVVs with nozzle diameters of 5.5 inches (14.0 Centimeters) and two RSVs with nozzle diameters of 3 inches (7.6 Centimeters) at the RPV head. There are two RRVs with nozzle diameters of 4.25 inches (10.8 Centimeters) at the upper RPV shell. The seven DBPB cases analyzed involved the inadvertent opening of one RVV, two RVVs, one RRV, two RRVs, one RRV plus one RVV, and one RSV and a CVCS pipe break. The outflow of the RVV, RRV, and RSV entering containment generates asymmetric loads for the CNV and RPV and the blowdown loads within the RPV. The asymmetric pressurization model and the blowdown model are both 180-degree symmetric models. In both models, mass was adjusted by combining point masses, distributed mass elements, and adjusted densities to account for the missing mass of the components that are excluded in the model. A fluid volume was created between the RPV and the CNV in the asymmetric pressurization model. In the blowdown model, a fluid volume was created within the RPV. The ANSYS acoustic fluid element was used to model the fluid. The CNV, RPV, lower RVI, and upper RVI are represented by ANSYS solid elements. The fluid acceleration and thrust force calculated by the NRELAP5 analysis were applied to the elements at the break location. ANSYS time history structural analysis was carried out. For each case analyzed, forces and moments at key structural cross sections and differential pressure loads were calculated. The NPM design used the bounding forces and moments in the seven cases. This section of the SER addresses the ANSYS modeling in TR-1016-51669, and SER Section 15.0.2 addresses the evaluation of the RELAP5 modeling.

TR-1016-51669, Revision 0, Section 3.2.3.2, “Flow Acceleration at Break Locations,” states that the acoustic elements represent the fluid inside the pipe in the ANSYS blowdown and asymmetric cavity pressurization models and that the flow acceleration was applied as a body force to the acoustic element nodes on the break face. The staff issued **RAI 502-9546,**

Question 03.09.02-78 (ADAMS Accession No. ML18243A502), asking the applicant to explain how it converted the flow acceleration to body force at the acoustic fluid element nodes on the break face to enable the staff to understand how the analysis model appropriately captured the input loading. In its response to **RAI 502-9546, Question 03.09.02-78** (ADAMS Accession No. ML18302A295), the applicant stated that it used the NRELAP5 results to generate the flow acceleration boundary condition and saved it in a text file for each break location. The applicant provided an example of ANSYS commands that read the acceleration data from the text file and convert the acceleration to body force on the break face. The applicant further stated that the NRELAP5 Heissdampf reactor benchmark study in TR-1016-51669 used the same methods and ANSYS commands. The calculated dynamic responses agree with the measured results as shown in TR-1016-51669. The staff finds the applicant's response acceptable. The agreement of the calculated dynamic response with the experimental results in the benchmark study demonstrates that the blowdown analysis appropriately captured the input loads. The staff is tracking **RAI 502-9564, Question 03.09.02-78,** as **Confirmatory Item 03.09.02-2** pending the applicant's change to TR-1016-51669.

The applicant provided bounding values of the calculated forces and moments of the seven cases of RPV valve openings and a CVCS pipe break in TR-1016-51669, Revision 0, Table 6-5, "Maximum Forces and Moments at Component Interfaces," and Table 6-6, "Maximum Forces and Moments on Containment Vessel, Reactor Pressure Vessel, Riser, and Core Barrel Assembly." However, the applicant did not provide a discussion. The staff issued **RAI 502-9546, Question 03.09.02-79** (ADAMS Accession No. ML18243A502), asking the applicant to identify the dominant valve opening cases that contribute to the bounding values of forces and moments in TR-1016-51669, Table 6-5 and Table 6-6, and explain why they are the dominant cases. The CVCS line terminates in the riser; a break in the CVCS line will result in a pressure wave traveling internally to the risers. The staff also asked the applicant to provide the maximum blowdown loads in the risers resulting from a CVCS line break. In its response to **RAI 502-9546, Question 03.09.02-79** (ADAMS Accession No. ML18302A295), the applicant stated that the case of two RVVs actuating simultaneously as a result of the high mass flow rates and correspondingly high fluid accelerations generated by this event causes most of the bounding forces and moments in the CNV, RPV, and RVIs. The applicant provided a markup of the revised TR-1016-51669, which would be Revision 1. The applicant added two tables that correlate the bounding forces and moments with the dominant-valve opening cases for the RPV, RVIs, and CNV. For the CVCS line break, the applicant added a table in the revised TR-106-51669 that provides the maximum forces and moments in the RPV and several riser locations generated by this case. The applicant stated that the maximum forces and moments at the RPV and riser locations of this case are bounded by the case of the two RVVs actuating simultaneously. The staff finds the applicant's response to be acceptable because the applicant has identified the dominant valve opening case for the bounding forces and moments at various locations of the CNV, RPV, and RVI in the revised TR-106-51669. The applicant also provided the maximum forces and moments in the RPV and riser locations caused by a CVCS line break. The staff is tracking **RAI 502-9564, Question 03.09.02-79,** as **Confirmatory Item 03.09.02-3** pending the applicant's incorporation of the changes to a subsequent revision of TR-106-51669.

<u>Stress Evaluation of a Reactor Vessel Internals Service Level D Faulted Condition</u>

DCA Part 2, Tier 2, Section 3.9.5, states that the RVI has the following major components:

- core support assembly (i.e., core barrel, lower core plate, reflector, upper core plate, and upper core support);

- lower riser assembly;
- upper riser assembly (i.e., upper riser, upper riser hanger support, and upper riser bellows);
- CRAGT, CRAGT support, control rod assembly card, and CRD shaft support; and
- SG tubes and tube supports, steam plenum, and feedwater plenum.

TR-0916-51502, Revision 0, does not provide a stress analysis or results for the RVI; therefore, the staff issued **RAI 202-8911, Question 03.09.02-18** (ADAMS Accession No. ML17237C062), asking the applicant to summarize the stress evaluation for the RVI components mentioned above and briefly describe the component structure modeling; input motion (time history or ISRS); major assumptions; acceptance criteria under a Service Level D Faulted Condition; fluid modeling; mass distribution; damping values; gap considerations; dominant modes and frequencies; seismic, DBPB, and MSPB/FWPB stress results; and an ASME B&PV Code, Section III, stress evaluation under a Service Level D Faulted Condition. The staff is tracking **RAI 202-8911, Question 03.09.02-18**, as **Open Item 03.09.02-11**.

<u>Stress Evaluation of Reactor Pressure Vessel Service Level D Faulted Condition</u>

In the 2017 audit report (ADAMS Accession No. ML18023A091) the NRC staff describes how NuScale evaluates the RPV primary stress in the Service Level D Faulted Condition. The stress report referenced on the audit report contains analysis assumptions, methodology, results, conclusions, and software documentation. The evaluation used the following load combination:

$$P + DW + EXT +/- SRSS(SSE + DBPB)$$

P, DW, EXT, SSE, and DBPB are defined as operating pressure, deadweight, external mechanical loads, safe-shutdown earthquake, and design-basis pipe break, respectively. The external mechanical load consists of RPV and CNV support reactions, RVI and CNV interface loads, scram loads, fuel assembly weights, and nozzle loads. The DBPB load includes load from the spurious actuation of RVVs, RSVs, and RRVs and from a CVCS pipe break. The staff noticed that the load combination does not include the blowdown loads of an MSPB and FWPB. During the 2017 audit, the applicant stated that it will perform an MSPB and FWPB blowdown analysis in the second quarter of 2018 and will conduct a confirmatory analysis to demonstrate that the blowdown loads of the MSPB and FWPB are bounded by the DBPB load. If blowdown loads are not bounded by the DBPB load, the applicant will reevaluate the RPV primary stress. In a memorandum dated September 20, 2018 (ADAMS Accession No. ML18264A273), the applicant stated that, because the MSPB and FWPB occur outside of containment, loads from asymmetric cavity pressurization do not apply to the RPV and a direct comparison of the loads caused by a design DBPB cannot be made. However, the applicant further stated that the MSPB and FWPB do affect the MS and feedwater nozzles and plenums. The applicant has updated the RPV primary stress analysis to include the loads at the feedwater nozzles and plenums. The calculation demonstrates that the inclusion of the MSPB and FWPB loads does not affect the conclusions of the analysis. The staff considers the blowdown loads acceptable because the applicant has updated the RPV primary stress analysis to include the loads of the MSPB and FWPB.

The RPV was evaluated in accordance with the stress limits specified in ASME B&PV Code, Section III, Appendix F, for the Service Level D Faulted Condition. Two ANSYS FE models were developed, one for the vessel top head (including a section of the pressurizer shell) and one for the vessel shell. The top head model is a 360-degree model that consists of 119,062 ANSYS solid elements (20 nodes per element). Vertical and circumferential displacements are fixed at the bottom of the model. Using the specified nozzle loads and seismic interface loads, a linear elastic analysis of the RPV top head was carried out to evaluate stress in the Service Level D Faulted Condition. Thirty-eight stress classification lines were defined at cross sections near the nozzle, the ligament between nozzles, the crown, and the knuckle of the top head. Using the ANSYS postprocess, stress linearization was performed to obtain membrane and bending stress intensities in these locations. The membrane stress intensities were classified as primary general membrane (P_m) and primary local membrane (P_L), in accordance with ASME B&PV Code, Section III, Table NB-3217-1, "Classification of Stress Intensity in Vessels for Some Typical Cases." The calculated stress intensities were compared with the ASME B&PV Code allowable stress limits of P_m, P_L, and P_L +/- P_b (P_b is "primary bending"). The analysis results indicate that the highest stress intensities occurred in the ligaments between the inner CRDM nozzles. The stresses were within the ASME B&PV Code allowable stress limits.

In addition, the vessel model is a 360-degree model that consists of the RPV below the top head. The model consists of 1,963,305 ANSYS solid elements. The model is constrained at the RPV upper support with vertical and tangential displacements fixed. Using the specified nozzle loads and interface loads, a linear elastic analysis was carried out. Forty-one stress classification lines were defined at cross sections near nozzles, ligaments between nozzles, the bottom head crown, the bottom head knuckle, and geometry discontinuity junctions. The analysis results indicate that the highest stress intensities occurred at RCS injection and discharge nozzles. The stresses are within the ASME B&PV Code allowable stress limits. In summary, the staff finds the stress analysis of the Service Level D acceptable because it demonstrates that the design of RPV meets the structural requirements of the ASME B&PV Code for Service Level D loads.

3.9.2.4.6 Correlations of RVI Vibration Tests with the Analytical Results

The applicant has not submitted correlations of RVI vibration test data and analytical results to date. DCA Part 2, Tier 2, Section 3.9.2.6, "Correlations of Reactor Internals Vibration Tests with the Analytical Results," states that the applicant will compare prototype tests to analytic results and that, if significant differences are observed, it will update the models and reconcile the differences. In addition, TR-0918-60894, Revision 0, does not provide the expected vibration levels, complete instrumentation, specifications, final test conditions, pretest predicted vibration and pressure levels, or acceptance criteria. The staff issued **RAI 427-9408, Question 03.09.02-77,** asking the applicant to provide this information. The staff is tracking **RAI 427-9408, Question 03.09.02-77,** as **Open Item 03.09.02-7**.

3.9.2.5 Combined License Information Items

DCA Part 2, Tier 2, Table 1.8-2, "Combined License Information Items," lists COL information item numbers and descriptions related to dynamic testing and analysis of SSCs from DCA Part 2, Tier 2, Section 3.9.2.

Table 3.9.2.5-1 NuScale COL Items for DCA Part 2, Tier 2, Section 3.9.2

Item No.	Description	DCA Part 2 Tier 2 Section
COL 3.9-1	A COL applicant that references the NuScale Power Plant design certification will provide the applicable test procedures prior to the start of testing and will submit the test and inspection results from the comprehensive vibration assessment program for the NuScale Power Module, in accordance with Regulatory Guide 1.20.	3.9
COL 3.9-10	A COL applicant that references the NuScale Power Plant design certification will verify that evaluations are performed during the detailed design of the MS lines utilizing acoustic resonance screening criteria and additional calculations as necessary (e.g., Strouhal number) to determine if there is a concern. The methodology contained in "NuScale Comprehensive Vibration Assessment Program Technical Report," TR-0716-50439 is acceptable for this purpose. The COL applicant will update Section 3.9.2.1.1.3 to describe the results of this evaluation.	3.9
COL 3.9-12	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific seismic analysis in accordance with Section 3.7.2.16. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the CSDRS, the standard design of NPM components will be shown to have appropriate margin or should be appropriately modified to accommodate the site specific demand.	3.9
COL 3.9-13	A COL applicant that references the NuScale Power Plant design certification will complete an assessment of piping systems inside the reactor building to determine the portions of piping to be tested for vibration and thermal expansion. The piping systems within the scope of this testing include ASME BPVC, Section III, Class 1, 2, and 3 piping systems, other high-energy piping systems inside Seismic Category I structures or those whose failure would reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and Seismic Category I portions of moderate-energy piping systems located outside of containment. The COL applicant may select the portions of piping in the NuScale design for which vibration testing is performed while considering the piping system design and analysis, including the vibration screening and analysis results and scope of testing as identified by the Comprehensive Vibration Assessment Program.	3.9

3.9.2.6 Conclusion

Until the open items are resolved, the staff cannot make a finding at this time.