

October 24, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 202 (eRAI No. 8911) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 202 (eRAI No. 8911)," dated August 25, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 8911:

- 03.09.02-19
- 03.09.02-20
- 03.09.02-21
- 03.09.02-22
- 03.09.02-23
- 03.09.02-25
- 03.09.02-26
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- 03.09.02-36
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- 03.09.02-39
- 03.09.02-40

- 03.09.02-41
- 03.09.02-42
- 03.09.02-44
- 03.09.02-47
- 03.09.02-48
- 03.09.02-49

The schedule for questions 03.09.02-18, 03.09.02-24, 03.09.02-43 and 03.09.02-46 were provided in an email to NRC (Greg Cranston) dated September 12, 2017. The response to question 03.09.02-45 will be provided by February 28, 2018.

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 202 (eRAI No. 8911). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The technical report TR-0916-51502, "NuScale Power Module Seismic Analysis" contained export controlled information. The markup pages in the enclosed RAI responses for TR-0916-51502 are therefore labeled "Export Controlled," although these markup pages do not contain any export controlled information. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,



Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8911, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8911, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-1017-56807

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8911, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 8911, nonproprietary

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-19

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. TR-0916-51502-P, Rev. 0, Section 4.0 states that three 3D ANSYS finite element models of the NuScale power modules (NPM) are developed. During normal operation, one of 12 NPMs is being refueled. Provide justification for not including in the seismic analysis for the scenarios in which the NPM is on the containment flange tool, reactor flange tool, in the dry dock, or in transit. Without addressing these scenarios, the staff cannot reach a safety finding. Include the requested information in the NPM Seismic Report.

NuScale Response:

The NuScale response to RAI No. 8769, Question 04.02-7 demonstrated that the seismic analysis performed for the NPM generated loads that bound those that result from other NPM placement locations. The locations evaluated in that assessment include the NPM placed in the containment flange tool, the reactor flange tool, and suspended by the reactor building crane, but did not include the temporary placement of the upper portion of the NPM in the dry dock.

During the refueling process, the NPM is disassembled and the lower reactor vessel assembly removed. The lower reactor vessel assembly contains the lower reactor vessel head, the core barrel, lower riser assembly, and the reactor fuel. The upper portion of the NPM is moved to the dry dock for service and inspections. The upper NPM is physically separated from the reactor fuel while in the dry dock by the pool wall that divides the dry dock from the refueling bay. Consequently, the upper module in this position does not meet the definition of Seismic Category I in Section 3.2.1.1 or the definition of Seismic Category II in Section 3.2.1.2, thus it is classified Seismic Category III per Section 3.2.1.3 while in the dry dock. Therefore, analysis of the module in the dry dock was not performed.



Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-20

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. In TR-0916-51502-P, Rev. 0, subsection 3.1.5, the applicant states that seismic analyses are performed by applying displacement time histories obtained from the Reactor Building soil structural interaction analysis to the NuScale power module (NPM) support locations and acceleration time histories to the reactor pool fluid surfaces in contact with the reactor pool floor and walls in the detailed 3D ANSYS Full Pool Model. In the detailed 3D ANSYS Full Pool Model, only one of the twelve NPMs is modeled. To account for the effect of NPMs that are not explicitly modeled, the containment vessel centerline acceleration time histories are applied to the surfaces of the fluid that would be in contact with the “missing” NPMs. Multiple seismic analyses are performed for each model. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Explain why displacement time histories are used for the boundary conditions of the NPM support locations (NPM skirt support and lugs) and not for the RP fluid surfaces in contact with the RP floor, walls, and the missing NPMs.
2. Explain what the multiple seismic analyses for each model are.

Include the requested information in the NPM Seismic Report.

NuScale Response:

At acoustic fluid element surfaces, the boundary conditions for transient dynamic analysis include either specified pressure or specified normal acceleration time histories. For the acoustic element surfaces, specified displacement is not an option provided by ANSYS. Acceleration time histories are applied on fluid boundaries at the reactor pool walls and floor and at fluid surfaces where NPM is not explicitly modeled. Zero pressure is specified on the top surface of the pool.

Displacement time histories are applied to structural supports where displacement degrees of



freedom exist (NPM skirt support and lugs). Displacement time histories used for the structural supports are obtained by double integration of the acceleration time histories.

When acceleration time histories taken from SASSI are double integrated and then input to the ANSYS analysis, drift or “baseline errors” may be introduced due to choice of the integration methods and initial conditions. The potential for drift is eliminated by introducing a small adjustment to the accelerations at the beginning of each record. The drift correction has negligible effect on the maximum response occurring during the strong motion portion of the excitation.

The use of enforced accelerations represents an alternate means to prescribe boundary conditions at the structural supports. The integration time step used in the analysis is sufficiently small that the two approaches produce approximately the same results.

Multiple seismic analyses cases were performed using the detailed 3D ANSYS models of the NPM and entire pool, as described in Section 8.0 of TR-0916-51502. The cases analyzed were:

1. NPM 1 and entire pool, Cracked Concrete Properties, Nominal NPM stiffness
2. NPM 1 and entire pool, Cracked Concrete Properties, 77% of Nominal NPM stiffness
3. NPM 1 and entire pool, Uncracked Concrete Properties, Nominal NPM stiffness
4. NPM 6 and entire pool, Cracked Concrete Properties, Nominal NPM stiffness
5. NPM 6 and entire pool, Cracked Concrete Properties, 77% of Nominal NPM stiffness
6. NPM 6 and entire pool, Uncracked Concrete Properties, Nominal NPM stiffness

The analysis cases were performed using inputs derived from the SASSI analysis of the RXB with soil profile 7 (Hard rock) and a CSDRS compatible control motion based on the Capitola recording.

These clarifications have been added to Section 3.1.5 and Section 8.0 of TR-0916-51502.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-21

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. TR-0916-51502-P, Rev. 0, Section 4.1 states that the NPM seismic model is created from five submodels: containment vessel and NPM pool bay, reactor pressure vessel, lower reactor vessel internals (RVI), upper RVI, and control rod drive mechanisms (CRDMs). Four of these submodels (not including the CRDMs submodels) were created in ANSYS. All five submodels are combined into a single model. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Does “submodel” mean “substructure” in ANSYS modelling technique?
2. Which computer code is used for CRDMs submodel?
3. Explain in detail how the five submodels are combined into a single ANSYS model.
4. Is master node of ANSYS substructuring technique used in combining submodels into a single master model? If so, describe the criteria in selection of the master nodes in the submodels and the main model.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: In TR-0916-51502-P, the term “submodel” does not refer to the term “substructure” as used by the ANSYS structural analysis program. In TR-0916-51502, the term “submodel” refers to assemblies or components that are separately defined in the ANSYS input. To form a single model, the submodels are connected by constraint equations, contact elements, or coupled degrees of freedom.

ANSYS also uses the term “submodeling” to describe a finite element technique used to obtain more accurate results in a particular region of a model, by using a refined model of the particular region. This usage of the term “submodel” is not the meaning intended in TR-0916-51502.



In the ANSYS documentation, "substructuring" refers to procedures that condense a group of finite elements into one element represented by a single mass, stiffness and damping matrix. The single-matrix element is called a "superelement." ANSYS superelements are also not used in TR-0916-51502.

Subquestion 2: The CRDM support frame and CRDM submodel described in TR-0916-51502 are generated by the ANSYS computer code. The ANSYS CRDM submodel is translated from a CRDM stress analysis model developed by the CRDM vendor using their proprietary structural computer code. Modal analyses were performed to verify the equivalence of the ANSYS to the CRDM vendor models.

Subquestion 3: Connections between or within the submodels are described in responses to the following RAI questions :

- RAI 8911 Question 03.09.02-23 (connection of mating surfaces of RPV and CNV),
- RAI 8911 Question 03.09.02-26 (connection of fuel assemblies to core plates),
- RAI 8911 Question 03.09.02-27 (connection between lower RVI and RPV),
- RAI 8911 Question 03.09.02-28 (connection upper support blocks to RPV and connection of reflector to core barrel),
- RAI 8911 Question 03.09.02-30 (connection of upper RVI and lower RVI and connections within the upper RVI, CRDMs, SG and RPV), and
- RAI 8911 Question 03.09.02-31 (connection within CRDM and CRDM supports).

Subquestion 4: The ANSYS substructuring technique and the concept of master nodes are not used in TR-0916-51502.

These clarifications have been added to Section 4.1 of TR-0916-51502.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-22

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. TR-0916-51502-P, Rev. 0, Section 4.1.1.3 states that the pool is assigned acoustic material properties. The description is insufficient for staff to reach a safety finding. Provide a complete list of the assigned acoustic material properties of the pool water. Include the requested information in the NPM Seismic Report.

NuScale Response:

The assigned material properties of the acoustic fluid elements used to represent pool water at 100 degrees F are provided in Table 4-4 of the NuScale Seismic Analysis technical report TR-0916-51502.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with this response.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-23

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. TR-0916-51502-P, Rev. 0, Section 4.1.2.2 (Reactor Pressure Vessel boundary conditions) states that the male end of the lower lateral reactor pressure vessel (RPV) support is coupled to the female end on the containment vessel (CNV) in the horizontal directions only. This is done using the pilot nodes for the two mating surfaces of the RPV/CNV alignment feature. The RPV upper support segments are connected to the CNV ledges in the vertical and circumferential directions only. This is done using pilot nodes scoped to the slots of the RPV support and holes of the CNV ledges. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Explain in detail the methodology in creating the pilot nodes.
2. Describe the function and input for the pilot nodes. Are there any constraint equations associated with the pilot nodes? If so, explain the constrain equations.
3. Discuss how the boundary conditions of the reactor pressure vessel are derived.
4. Can the RVP upper support segments uplift from the CNV ledges?

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestions 1 and 2: Pilot Nodes

Pilot nodes are manually created within ANSYS Mechanical as remote points. Faces of geometry are selected and the remote point is scoped to the faces. Scoping refers to the geometry over which a boundary condition is applied. ANSYS uses multipoint constraint (MPC) equations to generate a connection between the remote point and the meshed face to which the remote point is scoped. Those MPC equations comprise the function and input for the pilot nodes.

The remote points connecting the CNV and RPV submodels have options that specify them as deformable, meaning that the geometry to which each remote point is scoped is free to deform. A TARGE170 element is used at the remote point location, and CONTA174 elements at the scoped faces. For the TARGE170 element, all six degrees of freedom (DOF) are used in the multipoint constraint. The CONTA174 elements are constrained in the translational DOFs only. This deformable option is chosen because if the boundary conditions on the RPV were specified as rigid, the combined model would be constrained in an unrealistic way. That is, the remote points considered here are used to transmit loads between each submodel, and are abstractions that do not add inaccurate stiffness to the submodels.

Subquestion 3: RPV Boundary Conditions

In the seismic analysis presented in TR-0916-51502, loads are transmitted from one submodel to another by applying constraint equations to the pilot nodes (i.e., remote points) of two submodels in ANSYS Mechanical APDL. The constraint equations which couple pilot nodes from the RPV submodel to the CNV submodel define the boundary conditions of the RPV. At the bottom of the RPV, the pilot node associated with the RPV alignment feature and the pilot node associated with the CNV alignment feature are constrained to displace equally in the horizontal directions using the CE command in ANSYS Mechanical APDL. This is because the nominal design is a $\{\{ \}^{2(a),(c)}\}$ between the RPV and CNV alignment features in the cold condition. Modeling this horizontal gap as closed is a reasonable assumption. These nodes are not coupled in the vertical direction because the CNV alignment feature is designed to allow thermal expansion of the RPV alignment feature in the vertical direction without restraining motion.

The boundary conditions at the RPV Supports are governed by the nature of the connection with the CNV Ledges. The bolts which connect the RPV support slots to the CNV ledges are tensioned to accommodate thermal expansion of the RPV in the radial direction. Cylindrical coordinate systems are employed in order to capture this behavior using the CE command. Constraint equations are applied between the pilot nodes of the RPV supports and the CNV ledges in only the circumferential and vertical directions. This is because the rotation of the RPV about its vertical axis is not possible due to the four bolted connections securing it to the CNV ledges. By coupling the displacement of the pilot nodes of the RPV supports to the displacement of the pilot nodes of the CNV ledges in the circumferential direction, the connection is modeled appropriately. Likewise, the bolted connection at each of the four CNV ledges generates the need to couple the pilot nodes associated with the RPV supports and the CNV ledges in the vertical direction (i.e., there is no possibility for uplift).



Subquestion 4: RPV Supports Uplift Potential from CNV Ledges

As concluded above, the RPV upper support segments cannot uplift from the CNV ledges. The bolting for the RPV upper support ledge are classified as ASME Code Class 1 supports. These design requirements specify that the bolts are sized to withstand the maximum seismic loads.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-25

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. TR-0916-51502-P, Rev. 0, Section 4.1.3.1 states that the lower RVI submodel has its mass adjusted in a similar manner as the vessels. The core support mass is the sum of the following three RVI components: the core support, surveillance capsules, and core entrance flow plate. The negative mass adjustment associated with the core support was incorporated by reducing the density of the reflector material. This subsection provides a detailed description on mass adjustment in the lower RVI submodel. However, it lacks the description on mass adjustment to account for the fluid within the lower RVI, and the staff cannot reach a safety finding. The applicant is requested to provide a detailed description on how the fluid mass within the lower RVI is accounted for in the mass adjustment of the lower RVI submodel. Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

The mass adjustment that accounts for the fluid within the lower RVI is addressed in Section 4.1.8.1 of TR-0916-51502. The lower RVI fluid masses in Table 4-13 of that section are applied to the combined model by the APDL file that combines the five submodels. These fluid masses are applied either as horizontal or vertical point masses to applicable surfaces using MASS21 elements. Applicable surfaces are assigned as named selections within the submodels. The total mass to be applied to a region is divided by the number of nodes in the applicable named selection, and this mass is applied to each node in the named selection as a point mass. This is preferable to scoping a single point mass to a large surface, which can cause over-constraint due to the many constraint equations that are created. It is also preferable to the distributed mass option of surface elements, as those masses cannot be assigned independent directions. The directionality of the masses is assigned using the key options of the element type.

The horizontal fluid masses in question are listed as “Core + reflector channels,” “Lower riser,” and “Riser transition” in Table 4-13. These are applied to the inner surfaces of the reflector



blocks, the inner surfaces of the lower riser, and the inner surface of the riser transition, respectively. The vertical component of these fluid masses is contained within the "Main RCS Total (no PZR)" mass in Table 4-13. Half of this mass is applied using vertical point masses to the inside surface of the lower RPV head, and half is applied to the bottom of the pressurizer baffle plate.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-26

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. TR-0916-51502-P, Rev. 0, Section 4.1.3.2 states that within the lower RVI submodel is a model to represent the fuel assemblies. The model uses properties provided directly by the fuel vendor. The fuel model is a single fuel assembly stick model that consists of beam and spring elements. To account for all 37 fuel assemblies, the stiffnesses of the beams and springs were multiplied by Increasing the axial area of the beam elements by 37 also increases the mass by 37 for a total fuel mass. The combined mass and stiffness for a single fuel assembly is attached to the upper and lower core plates of the lower RVI submodel (fixed at the base and laterally/fully rotationally coupled on the top of the fuel assembly). The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Does the total mass include the fluid mass?
2. Does the spring element include only rotational spring? If yes, explain why only rotational spring is needed and translation spring is excluded.
3. Provide a FE mesh figure of the lower RVI model with the fuel assembly model attached and locations of the attachment.
4. Table 4-8 in section 4.1.3.2 depicts fuel beam element properties. Provide the formulas used in the calculations of the lower nozzle spring stiffness, fuel spring stiffness and upper nozzle spring stiffness. Provide key to the abbreviations BN, HMP, HTP1-4, and TN in Table 4-8?
5. Figure 4-6 in section 4.1.3.2 shows a simple sketch of the fuel assembly stick model with rotational springs attached. The applicant is requested to provide a figure of the full fuel assembly FE model with all the beams and spring elements. Also provide a sketch that uses traditional circle type legend to represent the rotational springs. If three rotational springs are needed for coupling in three directions, include all three springs in the sketch and provide detailed explanation of the sketch. The sketch should indicate the components and locations where the fuel assembly model is coupled to.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: The total mass does not include the fluid mass. The fluid mass determined in TR-0916-51502 (TR) Table 4-14 is added separately to the NPM full model.

Subquestion 2: The spring element only includes a rotational degree of freedom (DOF). A fuel assembly is primarily comprised of long, slender structures of a simple cross-section (i.e. fuel rods and guide tubes). The dynamic representation of such individual structures can be represented with simple linear models. The presence of the fuel rods and guide tubes in the fuel beam model are homogenized into a single beam representation, and are characterized by basic geometric and material properties such as cross-sectional area, moment of inertia, and elastic modulus of the material. However, the presence of spacer grids provides a coupling between individual fuel rods and guide tubes, which complicates the dynamic representation of the system. Under a lateral deflection of the fuel assembly, the fuel rods rotate within the spacer grid. Phenomenologically, the rotational springs in the fuel model represent the spacer grid stiffness resisting the rotation of the fuel rods. The rotational stiffness of the spacer grid cannot be characterized through an isolated test of the spacer grid component alone. Instead, the natural frequencies of the complete fuel assembly are characterized through testing. These measured frequencies are then used to benchmark the fuel beam model by tuning the rotational spring elements to match the experimentally determined behavior.

A translation spring is excluded from this model because it has no phenomenological connection to the dynamic behavior being modeled.

Subquestion 3: The FEM of the fuel core model and its connections to the lower RVI model is shown in Figure 4-7.

The Figure 4-7 has been added to the TR Section 4.1.3.2.

Subquestion 4: Key to abbreviations in Table 4-9 of the TR:

BN: Bottom Nozzle

HMP: HMP is a spacer grid product name. HMP grids are fabricated from nickel-alloys and are typically used as the lowermost spacer grid (i.e. lower end grid) in a fuel assembly design.

HTP1-4: HTP is a spacer grid product name. HTP grids are fabricated from zirconium alloys and are used at intermediate spacer grid locations on a fuel assembly. HTP spacer grids are also commonly used at the uppermost location (i.e. top end grid) of many fuel assemblies. The numbers 1 through 4 represent the spacer grid location, with 1 being the lowermost location, 2 and 3 being intermediate locations and 4 being the

uppermost location.

ISG: Intermediate Spacer Grid (HTP)

LEG: Lower End Grid (HMP)

TN: Top Nozzle

UEG: Upper End Grid (HTP)

Formulas used in the calculation of the Lower Nozzle Spring Stiffness, Fuel Spring Stiffness, Upper Nozzle Spring Stiffness:

Each of these springs represents the rotational stiffness at a given spacer grid location, as follows.

Lower Nozzle Spring Stiffness: Represents the rotational stiffness between the lower end grid (LEG) and the core plate. This rotational stiffness element applies to the HMP.

Fuel Spring Stiffness: Represents the rotational stiffness between the intermediate spacer grids (ISG) and the core plates. These rotational stiffness elements apply to HTP1, 2, and 3.

Upper Nozzle Spring Stiffness: Represents the rotational stiffness between the upper end grid (UEG) and the core plate. This rotational stiffness element applies to HTP4.

These spring stiffness values are not calculated, therefore formulas are not provided. The values are tuned based on a benchmark of the fuel beam model to experimentally measured frequency values for a single fuel assembly. See the response to Subquestion #2 above. The rotational stiffness for this single assembly model is multiplied by 37 in order to achieve a lumped representation of a full core of NuScale fuel.

Subquestion 5: The fuel core FEM beam and spring elements are shown in Figure 4-8. The sketch showing rotational springs is in Figure 4-9. These springs have rotational DOFs about two horizontal directions (X-axis and Z-axis). The beam elements have six DOFs at each node: translations in x, y, and z directions and rotations about the x, y, and z axes. The beam elements are homogenized representations of the entire fuel core cross-section, and have aggregate cross-sectional properties of the fuel rods, guide tubes, and the instrumentation tube. The top and bottom of the fuel are coupled to the upper core plate (UCP) and lower core plate (LCP) respectively, by defining rigid region (CERIG command), to represent top and bottom nozzle assemblies with rigid stiffnesses.

Figure 4-9 in TR-0916-51502 has been updated per this response. Figure 4-8 is added to the TR-0916-51502. The abbreviation for degree of freedom (DOF) has been added to Table 1-2.



Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-27

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. TR-0916-51502-P, Rev. 0, Section 4.1.3.3 (Lower Reactor Internals Boundary Conditions) states that the four lower core plate tabs on the lower RVI are coupled to the four lower core support blocks of the RPV submodel in the horizontal direction. For the vertical boundary condition, four spring elements are added between the four lower core plate tabs and the four lower core support blocks. These springs represent the Belleville washers at this connection. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Describe what are the ANSYS elements used in the coupling of the four RVI lower core plates tabs to the four lower core support blocks of the RPV and how the couplings are achieved in two separated submodels.
2. What are the ANSYS elements used for representing the Belleville washers?
3. List the spring constant of the spring element and explain how it is determined.
4. Provide a FE mesh figure that shows the locations of the couplings between the lower RVI and the RPV and the spring elements.
5. Are there any nonlinear effects such as gaps exist in the boundary conditions? If so, address them and explain how the nonlinear effects are considered in the analysis.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Overview

In the separate submodels for the Lower RVI and the RPV, pilot nodes are created using the methodology described in the response to RAI 8911 Question 03.09.02-23. The ANSYS elements used to vertically connect the four RVI lower core plate tab pilot nodes to the four lower core support block pilot nodes of the RPV are COMBIN14 elements. These nodes are

coupled in the radial and circumferential directions, as detailed in the response to RAI 8911 Question 03.09.02-23. To model the response of the Belleville washers, the COMBIN14 element receives an appropriate spring constant input. An FE mesh figure is presented below to show the location of the connection between a Lower RVI core plate tab and a core support block.

Subquestion 1: Coupling Submodels Using Pilot Nodes and the COMBIN14 Element

Pilot nodes are created within ANSYS Mechanical as remote points. See the response to RAI 8911 Question 03.09.02-23 for a description of the methodology for creating pilot nodes. The Belleville washers are represented by COMBIN14 elements. These elements have longitudinal or torsional capability in 1-D, 2-D, or 3-D applications. In this application, key options specify the Belleville washers as 1-D longitudinal springs in the vertical degree of freedom. The methodology for determining the spring constant input that the COMBIN14 element receives is discussed below.

Subquestion 2 and 3: Determination of the Belleville Washer Spring Constant

The Belleville washers are sized to tune the core's vertical natural frequency away from high vertical acceleration frequencies. To reduce the response of the core to those high vertical accelerations, which peak near 17 Hz, a target natural vertical frequency of 6 Hz was selected for the core. To achieve this target vertical frequency, the combined spring constant of the 10 Belleville washers acting in series at each core support block was calculated to be 106,800 lbf/in. Each washer was sized appropriately to avoid overstress, to avoid flattening out, and to keep the lower core plate tabs from touching the core support blocks during a seismic event. Additionally, since Belleville washers cannot carry loads in tension, three washers are included above each tab to prevent uplift and maintain the desired spring rate when the Belleville washers beneath each tab are no longer in compression. When acting in series, the three washers above each core plate tab also have a combined spring constant of 106,800 lbf/in. The three washers above each core plate tab are restrained vertically by the retaining nut above them.

Subquestion 4: FE Mesh Figure of Lower RVI and Core Support Block Coupling

Figure 4-10 is added to TR-0916-51502 Section 4.1.3.3. Figure 4-10 shows the meshed FE model with one element of the core support block made transparent for viewing the overlapping pilot nodes of the lower core plate tab and the core support block. These nodes overlap initially, but move relative to each other vertically once gravity and seismic loads are applied. The COMBIN14 element that models the Belleville washers spans these nodes.

Subquestion 5: Discussion of Nonlinear Effects



There are no nonlinear effects, such as gaps, in the connection between the Lower RVI core plate tabs and the core support blocks on the RPV because the Belleville washers are springs themselves represented by linear spring elements.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-28

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. TR-0916-51502-P, Rev. 0, Section 4.1.3.3 (Lower Reactor Internals Boundary Conditions) states that the upper support blocks of the lower RVI are connected to the inner walls of the core region of the RPV submodel. This is done using a no-separation contact between the four pairs of mating surfaces. Likewise, the reflector is connected to the inner surface of the core support barrel using a no-separation contact. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Describe the meaning of no-separation contact. Does it allow sliding between the mating surfaces? If so, what are the friction coefficients and how are they determined?
2. Which ANSYS elements are used for representing the no-separation contact condition? Describe the element and input data of the element. If the element represents a one-way compression-only one-way spring, describe the spring constant and how it is determined.
3. Indicating in Figure 4-5 where the no-separation contact surfaces are located.
4. What are the areas of the mating surfaces for the lower RVI and the reflector? Are the actual areas considered in the model?
5. Provide finite element mesh figures that show the locations of the no-separation contact.
6. Are there any nonlinear effects such as gaps exist in the boundary conditions. If so, address them and explain how the nonlinear effects are considered in the analysis.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: "No-separation contact" means contact detection points that are either initially inside the pinball region or that once involve contact, always attach to the target surface along the normal direction to the contact surface (sliding is permitted). No friction coefficient is assigned to the elements.

Subquestion 2: The no-separation contact is assigned using element types CONTA174 and TARGE170, which are a 3-D, 8-node Surface-to-Surface contact and 3-D Target segment, respectively. The main inputs include updating contact stiffness at each iteration, multipoint constraint (MPC) contact, and contact detection point on nodal point - normal to target surface, etc. These elements do not represent compression-only one-way springs.

Subquestion 3: The upper support blocks to RPV shell no-separation contact surfaces are indicated in Figure 4-6 (left hand side figure green areas).

The Figure 4-6 in TR-0916-51502 (TR) has been updated in accordance with this response.

Subquestion 4: The areas of the mating surfaces for the lower RVI and the reflector are indicated in Figure 4-6 (right hand side figure red lines). The actual areas are considered in the model.

Subquestion 5: The contact surface meshes are shown in Figure 4-11 and Figure 4-12.

TR-0916-51502 Section 4.1.3.3 has been updated to add Figure 4-11 and Figure 4-12.

Subquestion 6: Nonlinear effects are not considered for the radial gaps in the boundary conditions because the radial gaps are small (e.g. gap between reflector blocks and core barrel is 0.125 inch). Therefore, they can be modeled as linear supports. The contact behaviors are described in TR-0916-51502 Table C-1, Interfacing components, "Upper support blocks with the lower RPV shell" and "Reflector blocks with each other and the lower core plate," which show that only sliding needs to be considered for these surfaces in the NPM model.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-29

10 CFR 50, Appendix A, GDC 4 requires structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The Standard Review Plan (SRP) establishes criteria that the NRC considers acceptable to use in implementing the agency's regulations. SRP 3.9.2 states that the number of element is adequate when additional degrees of freedom do not result in more than a 10-percent increase in responses. TR-0916-51502-P, Rev. 0, Section 4.1.4.1 states that the upper RVI submodel geometry is based on the upper riser drawings. The computer-aided design model used to generate the drawings was defeatured and simplified in order to reduce the element count of the mesh. A cutaway view of the upper RVI submodel mesh is shown in Figure 4-7. The upper RVI is meshed using 8-node solid shell elements and 8-node solid elements.

The solid shell elements are used at any shell section where there is one element through the thickness. The solid elements are used at intersection regions of shells and where there is more than one element through the thickness. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Provide a discussion of how the SRP provision regarding the adequate number of elements is addressed considering the reduced element count of the mesh.
2. Provide a cutaway view of upper RVI before defeature and simplification and provide detailed description of the major components in the upper RVI including their dimensions and wall thickness.
3. Provide a detailed description of the upper RVI submodel mesh in Figure 4-7.
4. Explain any major components that are not included in the upper RVI submodel and provide justifications.
5. Provide a FE mesh figure of the Upper RVI that shows the locations of the 8-node shell elements and the 8-node solid elements.
6. The FE mesh in Fig. 4-7 appears to have beam elements. Are there any beam elements and spring elements in the upper RVI submodel?

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: This information is provided in the response to RAI 8911 Question 03.09.02-24.

Subquestion 2: The upper reactor vessel internals assembly is primarily composed of the upper riser shell. The upper riser shell is a long cylindrical structure that is approximately {{

}}^{2(a)(c)} tall and {{}}^{2(a)(c)} in diameter with {{}}^{2(a)(c)} wall thickness. The bottom of the upper riser shell is attached to a cone that connects the upper riser to the lower riser. The upper riser shell is bolted to the pressurizer baffle plate by the upper riser hanger ring. The upper riser hanger ring is a {{}}^{2(a)(c)} thick plate. The upper riser hanger braces connect the upper riser shell to the upper riser hanger ring.

Within the upper riser shell are five CRD shaft supports. These supports guide the CRDM shafts from the CRDMs to the fuel and provide guidance and support for the in-core instrumentation. As a secondary role, these supports also stiffen the upper riser shell. The members of these supports are approximately {{}}^{2(a)(c)} tall.

The upper riser bellows are between the upper riser shell and the upper riser cone section. The bellows allow for vertical thermal growth while limiting relative horizontal deflections between the upper riser and the lower riser. While the geometry of the bellows has not been explicitly modeled, its effect has been captured by coupling the upper riser and lower riser in the horizontal directions while the vertical direction is not coupled.

The upper riser geometry and general dimensions are shown in Figure 4-13.

The in-core instrumentation guide tubes, riser backing strips, and the RCS injection piping from the CVCS are not modeled. None of these structures significantly affect the stiffness of the upper riser assembly and therefore do not impact the analysis results.

The NPM Seismic Analysis technical report, TR-0916-51502, has been revised to include this additional technical discussion.

Subquestion 3: The upper RVI submodel is composed of the upper riser shell, upper CRDS supports, upper riser hanger ring, and upper riser hanger braces. The upper riser shell and upper riser hanger ring are meshed with SOLSH190 elements while the upper CRDS supports and upper riser hanger braces are meshed with SOLID185 elements. Figure 4-14 presents the mesh of the upper riser. SOLID185 elements are shown in red and SOLSH190 elements are shown in blue. One half of the horizontal steam generator mass is distributed to the upper riser shell using MASS21 elements. The MASS21 elements are integrally meshed to each node on the upper riser shell. See the response to RAI 8911 Question 03.09.02-32 for further details on the representation of the steam generator.



The NPM Seismic Analysis technical report TR-0916-51502, has been revised to include this additional technical discussion.

Subquestion 4: With the exception of the steam generator, the major components within the upper internals are modeled. See the response to RAI 8911 Question 03.09.02-32 for further details on the representation of the steam generator.

Subquestion 5: Figure 4-14 presents the FE mesh figure requested.

Subquestion 6: There are no beam or spring elements in the upper RVI submodel.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-30

10 CFR 50, Appendix A, GDC 4 requires structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The Standard Review Plan (SRP) establishes criteria that the NRC considers acceptable to use in implementing the agency's regulations. SRP 3.9.2 states that the number of element is adequate when additional degrees of freedom do not result in more than a 10-percent increase in responses. For boundaries conditions of the Upper RVI submodel, TR-0916-51502-P, Rev. 0, Section 4.1.4.2 states" the following:

The cone of the upper RVI is coupled to the cone of the lower RVI submodel in the horizontal directions only.

Eight rectangular contact surfaces on the upper RVI are coupled to the tips of the lower radial cantilever SG supports RPV submodel by coupling the eight pairs of pilot nodes in the radial direction only. Each pair of pilot nodes was created at the same location to avoid an inaccurate constraint.

Radial coupling is provided between the upper riser and the RPV using constraint equations. These represent the radial load transfer that occurs due to the stack-up of SG tube supports between the upper riser and RPV. The load transfer occurs along the height of the SG at the 8 support locations around the circumference.

The upper riser ring hole locations on the upper RVI are coupled to pin locations on the baffle plate of the RPV submodel. This is done by coupling the translational degrees of freedom on the eight pairs of pilot nodes.

The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Are there any nonlinear effects such as gaps existing in the couplings and connections of the CRDMs? If so, address them and explain how the nonlinear effects are considered in the analysis.

2. Explain in detail the physical connections between the upper RVI and the lower RVI. Provide figures to demonstrate the coupling. Explain how the couplings can be achieved in the models. Is the coupling in the horizontal direction only?
3. The Seismic Report has no information about modelling of the SG. Explain in detail of the physical connections between the upper RVI and SG supports. Provide figures to demonstrate coupling of the upper RVI with the lower radial cantilever SG supports RPV submodel. How far apart are the eight pairs of pilot nodes? Which NMP submodel contains the SG? Provide the FE model description that contains the SG.
4. Explain in detail the physical connections between the upper riser and the RPV. Where are the locations of radial coupling between the upper riser and the RPV? Provide sketches or figures to demonstrate the coupling. Provide the constraint equations and explain the reason not to use pilot node approach.
5. Explain in detail the physical connections between the upper RVI and the baffle plate of the RPV. Provide figures to demonstrate the coupling and locations of the pilot nodes.
6. Explain how nodes and elements are numbered in the upper RVI submodel, lower RVI submodel, and RPV submodel and how they are glued to a NPM full model.
7. It is confusing to use the term of submodel for upper RVI, lower RVI, and RPV. Submodel technique is generally used at regions with high stresses that need refined mesh. The displacement field calculated from a whole structure having a course mesh are used as input (boundary conditions) to the region of interest with refined mesh (i.e., submodel) to get the accurate highly-refined stress field. In the NPM modelling, it appears that all the submodels have course meshes and they are combined to a full NPM model. It is not consistent with the traditional submodelling technique. The deviations from the traditional submodelling technique should be explained at beginning of the NPM Seismic Report.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: Nonlinear effects are not included in the couplings and connections of the CRDMs. The couplings and connections of the CRDMs are addressed in the response for RAI 8911 Question 03.09.02-31 Subquestion #2. The control rod drive system (CRDS) and control rod assembly (CRA) are not explicitly modeled in the NPM model. The CRDS and CRA masses are included in the NPM model as presented in TR-0916-51502 (TR), Sections 4.1.8.3 and 4.1.8.4, respectively.

Subquestion 2: The physical connection between the upper RVI and the lower RVI is shown in the TR Figure C-18 and described in the TR Table C-1, Interfacing component, "Upper riser with lower riser." In the NPM model, shown in Figure 4-15, the pilot nodes of the upper riser cone and the lower riser cone are coupled using constraint equations. The couplings are assigned in two horizontal directions only.

Figure 4-15 has been added to the TR Section 4.1.4.2.



Subquestion 3: The response is provided in the response to RAI 8911 Question 03.09.02-32. The SG is not explicitly modeled in the 3D NPM model. Only the SG mass is included in the submodels of RPV and URVI.

Subquestion 4: The physical connections between the upper riser and the RPV are shown in Figure 4-16 and Figure 4-17. The support bar assemblies are not welded or bolted together but stacked in the radial direction. For the detailed seismic analysis with entire reactor pool and a single NPM (the Entire Pool model), constraint equations are assigned between the upper RVI and the RPV in the radial direction, as shown in Figure 4-18, to represent the radial load transfer that occurs due to the stack-up of SG tube supports between the upper riser and RPV. Using the pilot node approach for these couplings is not necessary because there are 8 sets of couplings in the circumferential direction and 6 sets in the vertical direction, so the moment created due to the skew between the coupling nodes is insignificant.

The Single Bay model, used for the generation of the simplified beam model shown in TR Section 6.4, does not have radial coupling between the upper riser and the RPV. Overall, the horizontal harmonic response of the Single Bay models with and without radial coupling between the upper riser and the RPV is not significantly different. The harmonic force amplitudes at key locations from these two models are compared in Figure 1 shown below.

{{

}}(2)(a)(c)



Figure 1 Single bay model reaction force amplitudes (loads in east-west direction)

Figure 4-16 through Figure 4-18 have been added to the TR Section 4.1.4.2. A description to distinguish various radial coupling applications between the models has been added to Section 4.1.4.2 as well.

Subquestion 5: The connections between the upper RVI and the baffle plate are shown in the TR Figures C-14 and Figure C-15, and described in TR Table C-1, Interfacing component, "Upper riser hanger ring with the PZR baffle plate." In the NPM model, coupling of the pilot nodes of the upper riser hanger ring and the RPV baffle plate is assigned in three translational DOFs, shown in Figure 4-19.

Figure 4-19 has been added to the TR Section 4.1.4.2.

Subquestion 6: The submodels are imported into the full model in the order of CNV, RPV, lower RVI, upper RVI, and CRDM. In ANSYS, every time a submodel is read to the full model, the node and element numbers in the existing full model are offset to higher numbers. A node number offset is applied to each submodel before combining it with the rest of the assembly. Coupling and boundary conditions are applied as needed.

Subquestion 7: Table 1-3 in the TR has been updated to include the definition for submodel as "The finite element model of a specific component (CNV, RPV, lower RVI, upper RVI, or CRDMs) in the NPM full model. Each submodel is created separately using either ANSYS Workbench or APDL code, and written out as a coded database file (.cdb). The submodels in this report are not the geometry regions with highest stresses that need a refined mesh as used in the submodeling technique described in the ANSYS manual." See the response to RAI 8911 Question 03.09.02-21, Subquestion 1.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-31

10 CFR 50, Appendix A, GDC 4 requires structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The Standard Review Plan (SRP) establishes criteria that the NRC considers acceptable to use in implementing the agency's regulations. The SRP 3.9.2 states that the applicant should consider any nonlinear effects in the analysis. In the CRDM Submodel, the applicant states in TR-0916-51502-P, Rev. 0, Section 4.1.5.1 that the CRDM is constrained to the RPV head by beam elements representing the CRDM nozzles. The nozzle elements share the two nodes through the thickness of the RPV head, and connect to the bottom of the CRDM. This couples the translational degrees of freedom, as well as rotation about the two horizontal axes. CRDM rotation about the vertical axis is constrained. The mid-heights of the CRDMs are coupled laterally to the CRDM support frame. The tops of the CRDMs are coupled laterally to the inside of the CNV top head opening. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Explain the modeling of the CRDM support frame and coupling between the CRDM support frame and the RPV head.
2. Are there any nonlinear effects such as gaps exist in the couplings and connections of the CRDMs? If so, address them and explain how the nonlinear effects are considered in the analysis. Alternatively, provide technical justification for how GDC 4 is being met if nonlinear effects have not been considered in the analysis.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: The CRDM pressure housing beam nodes are coupled in the lateral directions to the support grid. The CRDM support frame is also modeled with beam elements as shown in Figure 4-20 of TR-0916-51502. The cross-sections of these beam elements are either a rectangle (representation of the support plates) or hollow rectangles (other beams in the CRDM



support frame). The connection to the RPV head is facilitated by six degree-of-freedom target/contact pairs (bonded) to the proximal surfaces of the RPV head solid elements.

The Seismic Analysis TR-0916-51502 Section 4.1.2 has been updated to include this description.

Subquestion 2: No nonlinear effects are included in the linear analysis. The radial gaps between the CRDM latch housings $\{\{ \}^{2(a),(c)}$ and the seismic support plates $\{\{ \}^{2(a),(c)}$ are sized nominally to provide a tight fit in the cold assembly condition. Similarly, the tops of the CRDMs are connected laterally to the inside of the CNV top head opening, and the gaps are closed by the seismic support frame bolts. The CRDM pressure housings are welded to the RPV head nozzle safe ends.

TR-0916-51502 Section 4.1.5 has been updated to include this information.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-32

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Section 4.1.8.2 states that half of the steam generator (SG) horizontal mass is applied to the riser, and 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 description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Explain why there is no simplified SG submodel within the 3D NPM model.
2. Explain how the SG mass is distributed to the riser and RVP surfaces, by adjusting mass density of the shell elements or assigning mass elements at the nodes.
3. Discuss how the SG stiffness is considered in the NPM model.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: The steam generator assembly is comprised of two interwoven helical steam generators. The support structures are shared between each steam generator.

The steam generator assembly spans the annulus between the reactor pressure vessel and the upper riser. The steam generator tubes are supported by 21 sets of 8 tube support bar assemblies. The tube support bar assemblies are long with a relatively small cross section. These tube support bar assemblies hang from the T-shaped upper tube support bars on the integral steam plenum plate. The bottom ends of the tube support bar assemblies are supported by the lower tube support cantilevers. The lower tube support cantilevers provide circumferential restraint, and allow for vertical and radial thermal growth of the tubes.

The steam generator tubes rest on the tube support bar assemblies, which provide radial and vertical support but allow the tube to slide axially along the tube length. The tubes are

constrained at the feed and steam plenum tubesheets. The tube support bar assemblies are interconnected in the circumferential direction via the overhanging tabs from adjacent supports. This coupling effectively ties the tube support assemblies together in the circumferential direction.

The tube support bar assemblies also interface with each other through radial contact. This allows loads to be transmitted radially from the riser to the RPV shell. In the NPM model, the upper riser is radially coupled to the RPV shell at the tube support bar assembly interface locations to model this behavior (see the response to RAI 8911 Question 03.09.03-30, Subquestion 4, for additional discussion). However, the stiffness of the steam generator assembly in the other directions is inherently flexible and therefore, not modeled. The comparative stiffness of the reactor pressure vessel, including the steam generator properties beyond its mass contribution, and the radial coupling of the upper riser to RPV shell, do not have a significant effect on the results presented in TR-0916-51502.

The aspects of the steam generator assembly that are physically modeled are the lower tube support cantilevers and upper tube support bars on the integral steam plenum plate. The lower tube support cantilevers are attached to the reactor pressure vessel. The tip of the lower tube support cantilever is near, but not welded or bolted to, the upper riser lateral support pad (located on the upper riser at the elevation of the support cantilever). In the NPM model, coupling of the pilot nodes of the cantilever tip and support pad is assigned using constraint equations in the radial direction, shown in Figure 4-5. Each pair of pilot nodes was created at an identical location. The upper tube support bars are modeled as rectangular beams, integrally meshed directly to the integral steam plenum plate.

The NPM Seismic Analysis technical report, TR-0916-51502, has been revised to include this additional information.

Subquestion 2: The steam generator mass is distributed between the upper riser, reactor pressure vessel wall, and upper tube support bars. The vertical mass is evenly distributed between the bottom of the upper tube support bars. The horizontal mass is evenly distributed between the upper riser and reactor pressure vessel wall near the steam generator. The mass is applied to a named selection on each surface. The mass is divided by the number of nodes within the named selection, then applied as a point mass on each node. The directional mass is applied using MASS21 element key options within ANSYS. See Figure 4-23 for further information.

The NPM Seismic Analysis technical report, TR-0916-51502, has been revised to include this additional information.

Subquestion 3: This information is provided in the response to RAI 8911 Question 03.09.03-30, Subquestion 4.



Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-33

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Section 4.1.8.4 states that the mass of the control rod assemblies (CRA) is applied directionally. Half of the horizontal mass is applied to the CRA guide tube support plate, and the other half is applied to the upper core plate. The vertical mass is combined with the CRDS mass in the CRDM submodel. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Explain why half of the CRA horizontal mass is applied to the CRA guide tube support plate and the other half is applied to the upper core plate.
2. Are there any gaps between the CRA and the CRA guide tube support plate and the upper core plate. If so, provide a discussion of the gap sizes and how the gaps are modelled.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: In the normal operating position, the CRA are out of the core, and the rodlets contact the guide tube support cards. The hub on top of the CRAs contacts the CRDS alignment cones, structurally connected to the top of the lower riser by the CRA guide tube support plate. The load path for lateral seismic loads is through the upper core plate on the bottom, and to the support plate on top of the lower riser. These load paths are included in the model (the guide tubes and support cards are not). In the vertical direction, the CRDM drive shaft, with attached CRA, is supported during normal reactor operation (all rods out of the core) by the stationary grippers on top of the RPV (in the CRDM submodel), ignoring any friction on the CRA and drive shaft supports.

Subquestion 2: The radial gap between the CRDS alignment cone $\{\{ \}^{2(a),(c)}$ and CRA hub $\{\{ \}^{2(a),(c)}$ is controlled to provide a tight fit. The lowest CRA card is located $\{\{ \}^{2(a),(c)}$ above the top of the upper core plate (direct load path), and provides



for a closer fit to the control rods $\{\{ \}^{2(a),(c)}$. Therefore, gaps are not modeled.

NPM Seismic Analysis technical report TR-0916-51502 Section 4.1.8 has been updated to include this description.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-34

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Section 4.1.8.5 states that fluid coupling between the two concentric cylinders of the RPV and the RVI is accounted for by a method called Fourier Nodes. The method creates a set of constraint equations connecting the inner and outer surfaces of the annulus at several elevations along the annulus (23 locations for this model).

Only the first coupled mode ($M=1$, beam mode) between the RPV and RVI is considered because shell modes do not have a significant impact on the overall response of the NPM. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Provide mathematical terms of the beam-mode coupling mass matrices and the corresponding ANSYS element type.
2. Provide the procedure in setting up the Fourier nodes in the NPM model and a FE mesh figure showing the Fourier nodes.
3. Provide value of fluid density used in the mass matrix calculation.
4. Provide justifications that the shell modes of the RVI and RPV are not significant. In Figure 4-13 of EC-A023-3535, "RVI Turbulent Buffeting Degradation Evaluation," Rev. 0, the plot of the first mode of the lower riser appears to be a shell mode ($n=8$). Does this mode have any significant impact on the overall response of the NPM?

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: The ANSYS element type MATRIX27 with symmetric element matrices ($\text{keyopt}(2)=0$) is used for fluid coupling. For each element, 78 matrix constants are entered to the mass matrix, as shown in Equation 34-1 below. In Equation 34-1, F^a_{xn} and F^a_{zn} are the reaction forces at Fourier Node a which is coupled to the inner cylinder in the x-axis and z-axis, and F^b_{xn} and F^b_{zn} are the reaction forces at Fourier Node b which is coupled to the outer

cylinder in the x-axis and z-axis, respectively. For the beam mode, the hydrodynamic masses M_{11} , M_{12} , M_{21} , and M_{22} are expressed in Equation 34-2, which is from a ANSYS conference paper "Method for Hydrodynamic Coupling of Concentric Cylindrical Shells and Beams," presented at the 2004 International ANSYS Conference, Pittsburgh, PA, May 24-26, 2004.

$$\begin{bmatrix} F_{xn}^a \\ F_{zn}^a \\ F_{xn}^b \\ F_{zn}^b \end{bmatrix} = \begin{bmatrix} M_{11} & 0 & 0 & 0 & 0 & 0 & M_{12} & 0 & 0 & 0 & 0 & 0 \\ & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ & & M_{11} & 0 & 0 & 0 & 0 & 0 & M_{12} & 0 & 0 & 0 \\ & & & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ & & & & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ & & & & & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ & & & & & & M_{22} & 0 & 0 & 0 & 0 & 0 \\ & & & & & & & 0 & 0 & 0 & 0 & 0 \\ & & & & & & & & M_{22} & 0 & 0 & 0 \\ & & & & & & & & & 0 & 0 & 0 \\ & & & & & & & & & & 0 & 0 \\ & & & & & & & & & & & 0 \end{bmatrix} \begin{bmatrix} \ddot{u}_{xn}^a \\ \ddot{u}_{zn}^a \\ \ddot{u}_{xn}^b \\ \ddot{u}_{zn}^b \end{bmatrix} \quad \text{Equation 34-1}$$

$$\begin{aligned} M_{11} &= \frac{\pi \rho L a^2 (1 + r^2)}{(1 - r^2)} \\ M_{12} = M_{21} &= \frac{-2abr}{(1 - r^2)} \\ M_{22} &= \frac{\pi \rho L b^2 (1 + r^2)}{(1 - r^2)} \end{aligned} \quad \text{Equation 34-2}$$

where

M_{ii} = hydrodynamic mass ($i = 1$ or 2)

ρ = fluid density

L = axial length of the discretized segment

a = radius of the inner cylinder shell

b = radius of the outer cylinder shell

$r = a/b$

Subquestion 2: Fourier nodes are assigned at 23 elevations, as shown in Figure 4-25. At each elevation, two nodes are created at the center of the module as the Fourier nodes (identical location); one for RVI and the other for RPV. Each Fourier node is coupled to the wall inner surface in the radial direction at 16 locations along the circumference. In these constraint equations, the coefficients for the shell nodes are determined using the orthogonality of the Fourier series. Then, one MATRIX27 is created for each pair of Fourier nodes. Symmetric

matrices (keyopt(2)=0) and mass matrix (keyopt(3)=2) are assigned to the MATRIX27 element with the mass matrix constants assigned as real constants C1 through C78.

Figure 4-25 has been added to TR-0916-51502-P Rev.0 (TR) Section 4.1.8.5.

Subquestion 3: The fluid density is $\{\{ \}^{2(a)(c)}$ as listed in the TR Table 4-14 Region “Annular volume SG.”

Subquestion 4: The NPM 3-D model includes the effect of shell modes upon seismic response through solid and solid-shell representations of the RPV and RVI. For seismic analyses, the effect of fluid coupling of the RPV and RVI was considered as significant for only the beam mode. This is justified since the overall seismic inertial loading and response of the NPM is dominated by beam modes. Modal participation associated with local shell modes (determined without considering shell mode fluid coupling) is not significant. The frequency of shell modes such as the n=8 mode of the lower riser are above the range of significant seismic input and response is not sensitive to frequency changes due to fluid coupling. Therefore it was not necessary to include fluid coupling effects upon shell modes.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-35

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Section 6.1.4 states that the vertical response of the simplified CNV beam model is captured by a set of tuned point masses and springs that comprise the primary vertical modes from the 3D model. Table 6-2 contains the numerical values of a set of three tuned mass and spring constant in the dry CNV model and a set of four tuned mass and spring constant in the wet CNV model. No modal information about the primary vertical modes from the 3D model is given; therefore, the staff cannot reach a safety finding. The applicant is requested to explain what are the frequencies, types of the mode, and the associated NPM components that are represented by the set of the four tuned springs shown in Fig. 6-6 in the dry and wet CNV conditions. Which spring represents the fundamental dominant mode in the dry and wet conditions?

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

The vertical frequencies represented by the tuned springs are the ANSYS 3D model Y-Freq. values shown in Table 8-1 (dry model major modes) and Table 8-2 (wet model major modes). The frequencies of the tuned springs have been added to the NPM Seismic Analysis technical report TR-0916-51502 Table 6-2.

The dominant mode in the vertical direction is $\{\{ \quad \} \}^{2(a),(c)}$ Hz for the dry model (See TR Table 6-7 and Figure 6-16) and $\{\{ \quad \} \}^{2(a),(c)}$ Hz (See TR Figure 6-20) for the wet model. The dominant vertical modes consist of vertical translation of the RPV and RVI with small translation of the CNV. The predominant vertical deflection occurs where the RPV upper support segments are connected to the CNV ledges. For the wet model, the fundamental mode at $\{\{ \quad \} \}^{2(a),(c)}$ Hz (See TR Figure 6-19) is of lesser significance and consists of coupled vertical motion of fluid, CNV, RPV and RVI.



Higher modes near {{ }}^{2(a),(c)} Hz (dry model shown on TR Figure 6-17 and Figure 6-18; wet model shown on TR Figure 6-21 and Figure 6-22) consist primarily of the vertical response of the top support structure and CNV.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-36

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Section 6.1.5 states that the beam elements of the simplified RPV beam model are assigned as massless. Masses are assigned by point mass elements. The mass include everything inside the RPV. Torsional mass moment of inertia is also assigned in the mass element. The values of the mass and torsional mass moment of inertia are determined from the 3D detailed RPV ANSYS model shown in Figure 6-9 by slicing the model horizontally and extracting the properties for each section. Table 6-3 shows the masses and torsional mass moments of inertia at each node of the simplified RPV beam model. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Add the legend for the colors in the Fig. 6-9.
2. Add a column in Table 6-3 that depicts the names of the NPM subcomponents that creates the torsional mass moment of inertia and provide a discussion of the subcomponents.
3. Table 4-4 contains the total RPV mass calculated from the 3D detailed RPV ANSYS model. The total RPV mass by adding all the nodal masses in Table 6-3 for the simplified RPV beam model is considerably higher than the total RPV mass of the 3D RPV model. Explain the discrepancy.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: In TR-0916-51502, the colors in Figure 6-9 represent ANSYS element type numbers, and are not relevant for the model discussion. This is based on the ANSYS internal numbering system for groups of similar elements, and not indicative of the number in the element name (BEAM188 for example). The TR Figure 6-9 has been updated to depict the model using a uniform color.

Subquestion 2: A column with the names of individual NPM subcomponents that contribute



additional mass to the model has been added to TR Table 6-3. As indicated in Tuning Section 6.2, the missing mass of the RVI (not captured in the spring-mass elements shown in TR Figure 6-11) is proportioned to the RPV upper and lower RVI support elevation nodes. Masses of RPV attachments that are not explicitly modeled in the underlying 3D vessel model (TR Figure 6-9), such as cables, are indicated for individual nodes and considered for their contribution to the torsional mass moment of inertia.

Subquestion 3: The total RPV model mass of 484,519 pounds in TR Table 4-5 does not contain the RCS fluid mass at nominal reactor operating conditions (TR Section 4.1.8.1), nor half of the mass of the steam generator (TR Section 4.1.8.2), that are applied on the RPV and upper reactor vessel internals using APDL commands in the 'combine_submodels' input file (TR Section 4.1.4). See response to RAI 8911 Question 03.09.02-32.

The TR Section 6.1.5 has been updated to include this additional discussion.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-37

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Section 6.2 states that certain modes that are visible in the harmonic response of the RPV 3D model are associated primarily with the RVI, the reactor coolant, or other subcomponents of the RPV. The responses associated with these modes are difficult to capture in the simplified RPV beam model that does not include a separate set of beams to explicitly model the components. Therefore, the response is alternatively captured by employing a set of tuned point masses and springs to model the three most significant missing modes in the simplified RPV beam model. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. List all the visible modes in the harmonic response of the RPV 3D model, their frequency, type of the mode, and the component associated with the mode. Explain what is the reactor coolant mode and its significance.
2. What are the three most significant missing modes that are captured in the simplified RPV beam model? List the three modes in sequence of significance.
3. SRP 3.9.2 states that a sufficient number of modes should be included in the dynamic analysis to ensure participation of all significant modes. The criterion for sufficiency is that the inclusion of additional modes does not result in more than a 10-percent increase in responses. Discuss how this SRP criterion is met. Alternatively, provide technical justification for how 10 CFR 52.47 is being met if using some other methodology other than discussed in SRP 3.9.2.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: The significant modes visible in the RPV harmonic analysis are presented in Table 1, listed in order of significance. This table indicates the significant modes of the RPV 3D ANSYS model, where the significance is determined based on the reaction force in the

harmonic analysis at the RPV supports.

Reactor coolant modes are not indicated in Table 1. Distinct fluid modes are not represented since only inertial effects of the fluid are represented in the models.

Table 1 Significant RPV Modes

Frequency (Hz)	Components	Type of Mode	Note
{{	RPV	Bending	Captured in beam model before tuning
	Upper Riser/CRDM	Bending	Captured in beam model before tuning
	Fuel	Bending	Captured in beam model after tuning {{ }} ^{2(a)(c)}
	Lower Riser/Upper Riser/CRDM	Bending	Captured in beam model after tuning {{ }} ^{2(a)(c)}
	RPV	Bending	Captured in beam model after tuning {{ }} ^{2(a)(c)}
}} ^{2(a)(c)}	Lower Riser/CRDM/RPV	Bending	Captured in beam model after tuning {{ }} ^{2(a)(c)}

Notes:

(1) See Table 6-4 of TR-0916-51502

Subquestion 2: The three most significant or missing modes that are captured by the final tuning operation are reported in Table 6-4 of TR-0916-51502. The modes in Table 1 are listed in order of significance.

The CRDM, upper riser and lower riser modes presented in Table 1 differ slightly from those presented Table 6-4 of TR-0916-51502 ({{

}}^{2(a)(c)}, respectively). The values presented in Table 6-4 are the target frequencies for the spring-mass representation of the structure while the values presented in Table 1 are the actual frequencies from the 3D ANSYS model. When the spring-mass representation is added to the simplified beam model, the response of the structure is comparable to those presented in the first column of Table 1.

Subquestion 3: Section 6.5 of TR-0916-51502 presents a series of analyses for comparison of the simplified beam model and the ANSYS 3D model. This comparison ensures that the response of the simplified beam model closely matches that of the ANSYS 3D model. The tuning process described in Section 6.2 documents the process of adjusting the beam model to match the response of the ANSYS 3D model. The results presented in TR Section 6.5 show the two models respond in a similar manner. There are no significant modes of the ANSYS 3D



model within the frequencies of interest that are not captured by the simplified beam model.

The method by which the ANSYS 3D model satisfies the requirements of SRP 3.9.2 has been provided as part of the response to RAI 8911 Question 03.09.02-24.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-38

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. TR-0916-51502-P, Rev. 0, Section 7.2 states that from time history analyses of the NPM 3D model, time histories at locations of equipment supports within the NPM were calculated. Response spectra from the soil structure interaction (SSI) cases were then enveloped and broadened, according to ASCE 4-13 to give ISRS for use in design of the SSC supported within or directly on the NPM. NuScale FSAR Section 3.7.2.5 (Development of In-Structure Floor Response Spectra) states that development of ISRS follows the guidance in RG 1.122, Rev. 1, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components." The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. There are six 3D NPM seismic analysis runs. Each run will generate an ISRS in each direction at a given location. Explain how the six ISRS are enveloped at a given location for the design ISRS.
2. The use of ASCE 4-13 in the NPM ISRS generation is inconsistent with RG 1.122 which is utilized in the generation of the rest of ISRS within the RXB as stated in the NuScale FSAR Section 3.7.2.5. The major difference between the ASCE 4-13 and RG 1.122 is that ASCE 4-13 permits a 15% reduction in the narrow frequency peak amplitude if certain conditions are met. The 15% reduction of the narrow frequency peak amplitude is not consistent with RG 1.122 criteria. The use of 15% reduction of the narrow frequency peak amplitude may result in nonconservative seismic results. The applicant is requested to generate all design ISRS using the guidelines in RG 1.122 for seismic design of SSCs supported on the NPM or provide an alternative with justification.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: For each location, direction, and damping value, the response spectra were calculated for the six seismic analysis runs. Using each set of six response spectra, an envelope spectrum was constructed by finding the maximum of the six response values at each spectral frequency point. The envelope of the six spectra was then broadened by $\pm 15\%$ to produce the design ISRS.

Subquestion 2: The in-structure floor response spectra provided were generated using the guidance provided in RG 1.122, Rev. 1. Reduction of narrow frequency peak amplitudes was not performed.

This information has been included in TR-0916-51502 Section 7.2 and Section 8.0.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-39

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Section 8.0 states that the detailed 3D NPM model is analyzed using the Certified Seismic Design Response Spectra (CSDRS) compatible Capitola time histories. Outputs of the analysis include in-structure response spectra (ISRS), time history data, relative displacements, and forces and moments within the NPM. Since the high frequency CSDRS (CSDRS-HF) compatible time histories (i.e., the Lucerne time histories) are not considered in the analysis, the applicant is requested to clarify whether the NPM seismic analysis outputs based on the Capitola time histories are applicable to the seismic design of the NPM that is located at hard rocks sites with the CSDRS-HF. Without the clarification, the staff cannot reach a safety finding.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

As indicated in the response to RAI 8934 Question 03.07.02-13 subpart (d), NuScale is seeking design certification for the NPM considering seismic analysis for a single soil type (Soil Type 7) with single time history (CSDRS-compatible Capitola) input only. The analysis of the NPM demonstrates that the NPM design is acceptable and meets the requirements of 10 CFR Part 50, GDC 2, and 10 CFR Part 50, Appendix S at sites with characteristics consistent with these inputs.

The seismic analysis using a single CSDRS-based time-history and a single soil type input applies only to the NPM model. The seismic design of the reactor building (RXB) is based on analyses involving multiple time histories and soil types, which includes CSDRS-HF, as discussed in FSAR Section 3.7.2.4.



Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-40

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Section 8.2, Table 8-1 depicts the modal analysis results of the 3D detailed single bay dry NPM model. The table contains six dominant natural frequencies and their effective mass in the E-W (X) direction (longitudinal direction of the reactor building), four dominant natural frequencies in the N-S (Z) direction (transverse direction of the reactor building) and three dominant natural frequencies in the Y direction (vertical). No discussion on the modal results is given; therefore, the staff cannot reach a safety finding. The applicant is requested to provide the following information:

1. For each mode, identify the NPM component associated to that mode, type of the mode, percentage of the modal mass to the total mass and the total mass.
2. List all the modes up to 50 Hz in three directions from the NPM modal analysis. Provide the dominant modes of the CNV, RPV, lower RVI, upper RVI, CRDM, upper and lower core plate, core barrel, and SG.
3. In the N-S direction, mode 3 is the dominant mode with the highest effective mass. Provide an explanation why mode 3 is the dominant mode instead of the typical mode 1.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: Columns for the major NPM subcomponents responding at this frequency, types of mode, total mass fraction, and the total effective mass above a lowered mass participation cutoff of 0.1% are added to TR-0916-51502 Table 8-1. Note that the bending modes of the major vessels include shell deformation.

Subquestion 2: Modes with mass participation of 0.1% or higher up to 50 Hz were added to TR Table 8-1. The 40 tabulated modes are a subset of 115 modes up to 50 Hz. The steam generators are not explicitly modeled; instead, the mass of the steam generators is applied as directional surface mass elements (TR Section 4.1.8.2), and shows up in the modal response of



RPV and upper RVI. Refer to response to RAI 8911 Question 03.09.02-32.

Subquestion 3: The first two modes are fuel modes with low overall mass participation. The third mode at {{ }}^{2(a)(c)} Hz is the most significant, as it involves a primary CNV bending/shell mode with major RPV/internals displacement.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-41

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Section 8.2, Table 8-2 depicts the modal analysis results of the 3D detailed single bay wet NPM model. The table contains six dominant natural frequencies and their effective mass in the E-W (X) direction (longitudinal direction of the reactor building), three dominant natural frequencies in the N-S (Z) direction (transverse direction of the reactor building) and four dominant natural frequencies in the vertical direction. No discussion on the model results is given; therefore, the staff cannot reach a safety finding. The applicant is requested to provide the following information:

1. For each mode, identify the NPM component associated to that mode, type of the mode, percentage of the modal mass to the total mass. Also provide the total mass for all the modes.
2. Mode 1 in the E-W direction is the fuel assembly first mode (see Table 4-9). The fuel model is attached to the upper and lower core plates of the lower RVI submodel which is part of the 3D detailed NPM model. This mode should be present in the modal analysis of the dry and wet 3D NPM models in both the E-W and N-S directions. Explain the reasons why this fuel assembly mode is missing in the dry NPM modal analysis results in Table 8-1 and the N-S direction of the wet NPM modal analysis results in Table 8-2.
3. List all the modes up to 50 Hz in the three directions from the NPM modal analysis. What are the dominant modes of the CNV, RPV, lower RVI, upper RVI, CRDM, upper and lower core plates, core barrel, and SG.

Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

Subquestion 1: Columns for the major NPM subcomponents responding at this frequency, types of mode, total mass fraction, and the total effective mass above a lowered mass participation



cutoff of 0.1% are added to TR-0916-51502 Table 8-2. The total mass (static) of the model is $\{ \{ \}^{2(a),(c)}$ is the pool bay water mass. The sum of the percentage of modal mass to total mass in the horizontal directions exceeds 100% because of the effect of the confined hydrodynamic mass (see TR Section 6.6.5.2). Note that the bending modes of the major vessels include shell deformation.

Subquestion 2: The fuel assembly first mode was not shown in the other directions due to its relatively lower significance. The first and second bending modes have been added to TR Table 8-1 (see response to RAI 8911 Question 03.09.02-40) and Table 8-2 of TR-0916-51502.

Subquestion 3: Modes with mass participation of 0.1% or higher up to 50 Hz were added to Table 8-2 of TR-0916-51502, plus the second fuel mode at $\{ \{ \}^{2(a),(c)}$ Hz. The 34 tabulated significant modes are a subset of 121 modes up to 50 Hz. The modes not presented in this table have low mass participation and would not contribute to the overall response of the structure. The upper and lower core plates as well as the core barrel are included in the Lower Reactor Vessel Internals (LRVI) component and participate primarily in the vertical Y-direction.

The mass of the steam generators is applied as directional surface mass elements on the RPV and URVI (TR Section 4.1.8.2), and shows up in the modal response of RPV and upper RVI. Refer to response to RAI 8911 Question 03.09.02-32 for further details of the steam generator modeling.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-42

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Sections 8.4.2.3 and 8.4.2.4 state that maximum seismic forces and moments are generated for 22 component sections within the NPM components, including various elevations of the RPV, CNV, and RVI. The upper and lower risers and core barrel in the 3D NPM model are modelled by 16 shell elements in the circumferential direction. The description is insufficient for staff to reach a safety finding.

Explain how the maximum forces and moments in a cross section are determined from the 16 shell elements. Also explain how the maximum forces and moments in a cross section are determined from the six NPM seismic analysis runs. Include the requested information in the NPM Seismic Report or in separate reports.

NuScale Response:

The 3D NPM model is a “global” model that is used to compute internal load distributions, reaction forces and accelerations that are used to define the seismic loading applied for refined stress analysis of individual components using “local” finite-element analysis or classical methods. For this purpose, the resultant internal forces and moments acting upon cross sections across shell structures or upon interfaces are evaluated as follows:

- a) Nodal forces and moments associated with the elements adjacent to a cross section or interface are calculated for all time points.
 - b) Elements on one side of the cross section cut or interface are selected for following steps. The associated nodes on the cross section or interface are selected.
 - c) Forces and moments acting on the selected set of nodes from the selected elements are summed about a point at the centerline of the cross section or interface. Only forces and moments acting on the selected nodes and elements contribute to the resultant. For each time point, the resultant three force components and three moment components
-



are stored.

d) For each force and moment component direction, the maximum absolute value is determined. Maximum forces and moments are summarized in a seismic loading specification for use in subsequent analysis. Note that the maximum values may occur at different times and from different NPM seismic analysis runs. Time histories of each force and moment component may be used for detailed analysis when necessary.

These clarifications have been added to Sections 8.4.2.3 and 8.4.2.4 of TR-0916-51502.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with the response to question 03.09.02-44.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-44

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components (SSCs) sufficient to permit understanding of the system designs. TR-0916-51502-P, Rev. 0, Appendix B presents the representative ISRS plots in the three directions of the CNV lugs, CNV top head, RPV top head, lower core plate, upper core plate, and steam generator top and bottom sections. The description is insufficient for staff to reach a safety finding. The applicant is requested to provide the following information:

1. Provide a discussion of the mode and frequency associated with the two spectral peaks in the vertical ISRS of the lower and upper core plates shown in Figure B-18 and B-21, respectively. Explain why the frequency of the first peak (at about 5.5 Hz) is not included in Table 8-2 entitled "Model Analysis Results for the Single Bay Wet NPM model". Table 8-2 shows that the lowest vertical NPM mode much higher than 5.5 Hz.
2. Identify the mode and frequency associated of the spectral peaks in the East-West and North-South ISRS of the lower core plate. In Fig. B-17 (N-S ISRS of lower core plate), there is a noticeable spectral peak at frequency about 70 Hz. Provide a discussion of this mode and include this mode in Table 8-2.
3. Provide a discussion of the mode and frequency associated with the two spectral peaks in the vertical ISRS of the top and lower sections of the steam generator shown in Figure B-24 and B-27, respectively. Explain which peak is associated with the steam generator.
4. Provide a discussion of the mode and frequency associated with the two spectral peaks in the East-West (X) and North-South (Z) ISRS of the top and lower sections of the steam generator shown in Figure B-22, B-23, B-25 and B-26, respectively. Explain which peak is associated with the steam generator. Explain why the north-south ISRS at the bottom steam generator section (Fig. B-26) has much higher spectral peaks than those at the top steam generator section in Fig. B-23 (e.g., for 5% damp ISRS, 18g vs. 6g). Explain why the frequency of the second peak in the East-West ISRS at the top steam generator section (about 11 Hz shown in Fig. B-22) is different from the frequency of the second peak of the ISRS at the bottom of steam generator section (about 16 Hz shown in Fig. B-25). Provide a table and discussion of the dominant frequencies of the steam generator in the three directions and their mode shapes.
5. The NPM has two steam generators. Does the SG ISRS presented in Appendix B envelop both steam generators? If so, discuss the enveloping process.

6. One of spectral peak of the North-South ISRS of the CNV lugs (Figs. B-2, B-5, and B-8), containment vessel top head (Fig. B-11), and reactor vessel top head (Fig. B-14) occurs at frequency is about 7 Hz. This is inconsistent with Table 8-2 (Model Analysis Results for the Single Bay Wet NPM Model) which shows a different frequency for the lowest NPM mode in North-South direction. Explain the discrepancy.

Include the requested information in the NPM Seismic Report.

NuScale Response:

Subquestion 1: The first significant mode in the vertical direction (i.e. the mode with the largest mass participation in the vertical direction) of the NPM wet model is $\{\{ \quad \} \}^{2(a),(c)}$. This mode is shown in the technical report TR-0916-51502 (TR), Figure 6-19 and represents the fundamental vertical mode of the RPV, with pinned support of the CNV at the pool floor. The frequency of this mode was determined to be close to the significant vertical SSI frequencies of the building on soil profile 7.

The NPM seismic analysis with the reactor pool and a single NPM, included Belleville washers below the core support assembly, as described in TR Section 4.1.3.3. The Belleville washers decouple the core barrel vertical response from that of the supporting structures. The observed peak (between $\{\{ \quad \} \}^{2(a),(c)}$) in the ISRS in TR Figures B-18 and B-21, is due to the first major vertical frequency ($\{\{ \quad \} \}^{2(a),(c)}$) of the decoupled core support assembly, as shown in TR Figure 3-4. The core support structure mode has small mass participation and it is decoupled from the supporting structures. A direct comparison of the harmonic responses of the skirt vertical reaction force for Rayleigh damping (TR Figure 6-24) demonstrates the small contribution of the decoupled core support assembly to the overall NPM vertical response (peak at $\{\{ \quad \} \}^{2(a),(c)}$). The differences in the other (lateral) directions are negligible, and not displayed. TR Figure 3-3 shows the vertical response for the documented model, and TR Figure 3-4 the vertical response of the decoupled model. The broadening of the second significant vertical mode at $\{\{ \quad \} \}^{2(a),(c)}$ to two separate peaks at $\{\{ \quad \} \}^{2(a),(c)}$ is discussed in the following paragraph.

The second significant vertical mode of the decoupled model ($\{\{ \quad \} \}^{2(a),(c)}$, shown in the middle of TR Figure 3-4) has the highest mass participation and corresponds closely to the first vertical mode of the NPM wet model without Belleville springs ($\{\{ \quad \} \}^{2(a),(c)}$, shown in TR Figure 6-19. The third significant vertical mode of the decoupled model ($\{\{ \quad \} \}^{2(a),(c)}$, on the right of TR Figure 3-4) corresponds to the second significant vertical mode of the NPM wet model ($\{\{ \quad \} \}^{2(a),(c)}$, shown in TR Figure 6-20. Overall, the vertical modal response of the decoupled model is not significantly different from the NPM wet model without Belleville

washers ($\{\{ \} \}^{2(a),(c)}$ vertical mass participation in the $\{\{ \} \}^{2(a),(c)}$ frequency range), that was used for the generation of the simplified beam model shown in TR Section 6.4. The TR Figure 3-2 demonstrates the seismic model generation flowsheet.

This explanation has been added to the TR-0916-51502, Section 3.1.

Subquestion 2: The peak in TR Figure B-17 actually extends across a range of frequencies, from about $\{\{ \} \}^{2(a),(c)}$. Notable major NPM lateral frequencies in that range are at $\{\{ \} \}^{2(a),(c)}$ (with non-zero mass participation ratios in the lateral directions). The displacement modal responses of the lower vessel region at $\{\{ \} \}^{2(a),(c)}$ (E-W) and $\{\{ \} \}^{2(a),(c)}$ (N-S) are shown in the following Figure 44-4, with the displacement scale increased for better illustration. Significant modes up to 50 Hz have been added to the TR Tables 8-1 and 8-2, and an explanation added to TR Section 8.2.

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$\{\{ \} \}^{2(a),(c)}$

Figure 44-4 Wet model, lower vessel region modes at $\{\{ \} \}^{2(a),(c)}$ (N-S)

Subquestion 3: The peak in TR Figures B-24 and B-27 extends across a range of frequencies, from about $\{\{ \} \}^{2(a),(c)}$. The major NPM vertical frequencies in that range are at $\{\{ \} \}^{2(a),(c)}$ (TR Table 8-2). The steam generators are not included in the model as structural elements; however, the mass of the steam generators is applied as directional surface mass elements on the RPV and upper RVI (TR Section 4.1.8.2), and shows

up in the modal response of RPV and upper RVI. Refer to response to RAI 8911 Question 03.09.02-32 for further details of the steam generator modeling. This represents the floating vertical connection at the bottom cantilever interface, as the upper support bars carry all the vertical loads and the lower cantilever bars have gaps to allow for thermal expansion and do not carry vertical loads. Clarification has been added to the TR Section 8.2 that the “steam generator section” elevations in TR Figure A-2 (corresponding to TR Figures B-24 and B-27) do not represent horizontal SG mass application locations. The actual support cantilevers are modeled, as shown in Figure 44-5.

Subquestion 4: The peak in question for TR Figures B-23, B-25 and B-26 extends across a range of frequencies, from about $\{\{ \}^{2(a),(c)}$. The RPV is supported off the CNV by the RPV support ledges near the elevation of the top SG supports (TR Figure 2-7). The bottom SG supports are further away and experience a higher lateral acceleration than at the top. The dominant NPM lateral frequencies (bending modes) in that range are at $\{\{ \}^{2(a),(c)}$ (TR Table 8-2). The displacement of selected components for these two modes is shown in Figure 44-5, with the displacement scale increased for better illustration. The yellow contour color of the bottom SG supports indicates greater displacement. The steam generators are not included in the model; the mass of the steam generators is applied as directional surface mass elements, and shows up in the modal response of the RPV and upper RVI. Note that the SG support points for ISRS generation are located on the RPV (TR Figure A-6). The associated significant modes have been added to the TR Tables 8-1 and 8-2 and explanation has been added to TR Section 8.2.

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$\}^{2(a),(c)}$

Figure 44-5 Wet model, upper internals region modes at $\{\{$

$\}^{2(a),(c)}$ (N-S)

The second listed X-Frequency (E-W) peak in TR Table 8-2 at $\{\{ \quad \} \}^{2(a),(c)}$ is associated with a dominant RVI bending mode for ~17% total mass participation (of NPM and pool bay mass). The Z-Frequency (N-S) peak at $\{\{ \quad \} \}^{2(a),(c)}$ represents the major NPM bending mode with large mass participation. The displacement of selected components for these two modes is shown in the following Figure 44-6, with the displacement scale increased for better illustration. Associated significant modes have been added to TR Table 8-1 and Table 8-2, and explanation of the modal behavior to TR Section 8.2.

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$\} \}^{2(a),(c)}$

Figure 44-6 Wet model, upper reactor vessel support region modes at $\{\{ \quad \} \}^{2(a),(c)}$ (E-W)

Subquestion 5: The SG ISRS presented in the TR Appendix B envelops both steam generators. The actual configuration of the helical tube columns of the two SGs form an intertwined bundle of tubes around the upper riser assembly, with a total of four feed plena and four steam plena located 90 degrees apart around the RPV (shown in TR Figure 5.4-3). Therefore, the simplified SG modeling detail in the NPM seismic model is adequate for the overall NPM seismic response analysis. Subsequent component qualification remains unaffected. Clarification has been added to TR Section 4.1.8.2.

Subquestion 6: The lateral peak of concern in TR Figures B-2, B-5, B-8, B-11 and B-14 extends across a range of frequencies, from $\{\{ \quad \} \}^{2(a),(c)}$. Notable major NPM lateral frequencies around that range are at $\{\{ \quad \} \}^{2(a),(c)}$ (E-W, for ~67% total mass participation, see TR Figure 8-1) and $\{\{ \quad \} \}^{2(a),(c)}$ (N-S, for ~27% total mass participation, see TR



Figure 8-2), both representing the first NPM bending mode in the respective directions. The peak centered at $\{\{ \quad \} \}^{2(a),(c)}$ is the result of $\pm 15\%$ broadening of the surrounding peaks (see updated example Figure 8-9 for the 4% damping curve in TR Figure B-11). The types of modes, fraction of total mass, and the total mass have been added to TR Tables 8-1 and 8-2.

Impact on DCA:

The technical report TR-0916-51502 has been revised as described in the response above and as shown in the markup provided with in this response.

Table 1-1 Components supported by the NuScale Power Module

Reactor pressure vessel	Upper reactor vessel internals
Containment vessel	Lower reactor vessel internals
Containment vessel supports	Piping and valves
Steam generators	Control rod drive system
Pressurizer heater assemblies	Instrumentation and controls
Top auxiliary mechanical access structure	

The methodology described in this section is also used to determine core plate motion time histories required for seismic analysis of the fuel.

1.3 Abbreviations

Note: if the NRC and NuScale acronyms or abbreviations differ, the project acronyms and abbreviations shall be followed.

Table 1-2 Abbreviations

Term	Definition
APDL	ANSYS parametric design language
ASCE	American Society of Civil Engineers
CNV	containment vessel
CRA	control rod assembly
CRDM	control rod drive mechanism
CRDS	control rod drive shaft system
CSA	core support assembly
CSDRS	certified seismic design response spectra
DHRS	decay heat removal system
DOF	degree of freedom
FSAR	Final Safety Analysis Report
FSI	fluid-structure interaction
FW	feedwater
IEEE	Institute of Electrical and Electronics Engineers
ISRS	in-structure response spectra
LWR	light water reactor
NPM	NuScale Power Module
NRC	U.S. Nuclear Regulatory Commission
PZR	pressurizer
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide

Table 1-3 Definitions

Term	Definition
Operating basis earthquake	The vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The operating basis earthquake ground motion is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input.
Safe shutdown earthquake	The vibratory ground motion for which certain structures, systems, and components must be designed to remain functional.
<u>Submodel</u>	<u>The finite element model of a specific component (CNV, RPV, lower RVI, upper RVI, or CRDMs) in the NPM full model. Each submodel is created separately using either ANSYS Workbench or APDL code, and written out as a coded database file (.cdb). The submodels in this report are not the geometry regions with highest stresses that need a refined mesh as used in the submodeling technique described in the ANSYS manual.</u>

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}}^{2(a),(c)ECI}

Figure 2-6 Core support assembly

3.1 Summary of Analysis Steps

Each of the steps outlined in Section 3.0 are discussed below in more detail.

3.1.1 Detailed 3D Model of the NuScale Power Module in ANSYS

The initial step of the seismic design process is creation of a detailed 3D finite element NPM model, including the fluid volume representing a single RP bay (i.e. “single bay model”). In this step, the dynamic properties of the model are calculated by performing modal, harmonic and benchmark transient analyses. The analytical results are then used as a basis for the development of a simplified NPM beam model that is dynamically similar to the detailed 3D model. From modal analysis, the major modal frequencies and associated effective masses are determined. From the harmonic analysis, reaction force vs. frequency plots are generated at the NPM supports. From the benchmark transient analyses, a set of reaction force vs. time plots are generated at the NPM supports. The analyses described can also be performed for submodels (e.g., CNV only, RPV only) that are extracted from the complete detailed 3D finite element model. These submodels facilitate the independent creation of separate simplified models that can then be combined, thus simplifying the tuning process described in Section 3.1.2. The detailed 3D model of the NPM is further described in Section 4.0.

The NPM seismic analysis with the reactor pool and a single NPM includes Belleville washers below the core support assembly, as described in Section 4.1.3.3. The Belleville washers decouple the core barrel vertical response from that of the supporting structures. The observed peak (between $\{\{ \dots \}^{2(a),(c)}$ in the ISRS in Figures B-18 and B-21, is due to the first major vertical frequency ($\{\{ \dots \}^{2(a),(c)}$) of the decoupled core support assembly, as shown in Figure 3-4. The core support structure mode has small mass participation and it is decoupled from the supporting structures. A direct comparison of the harmonic responses of the skirt vertical reaction force for Rayleigh damping (Figure 6-24) demonstrates the small contribution of the decoupled core support assembly to the overall NPM vertical response (peak at $\{\{ \dots \}^{2(a),(c)}$). The differences in the other (lateral) directions are negligible and not displayed. Figure 3-3 shows the vertical response for the documented model, and Figure 3-4 the vertical response of the decoupled model. Figure 3-5 illustrates the mode shapes of the first three significant vertical modes at $\{\{ \dots \}^{2(a),(c)}$ of the decoupled model in Figure 3-4. The broadening of the second significant vertical mode of the documented model at $\{\{ \dots \}^{2(a),(c)}$ to two separate peaks at $\{\{ \dots \}^{2(a),(c)}$ in the decoupled model is discussed in the following paragraph.

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112(a),(c)

Figure 3-3 Vertical Response for the Model

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11^{2(a),(c)}

Figure 3-4 Vertical Response of the Decoupled Model

$\}}^{2(a),(c)}$

Figure 3-5 First Three Significant Modes of the Decoupled Model

The second significant vertical mode of the decoupled model ($\}}^{2(a),(c)}$, shown in Figure 3-4) has the highest mass participation and corresponds closely to the first vertical mode of the NPM wet model without Belleville springs ($\}}^{2(a),(c)}$, shown in Figure 6-19. The third significant vertical mode of the decoupled model ($\}}^{2(a),(c)}$, on the right of Figure 3-4) corresponds to the second significant vertical mode of the NPM wet model ($\}}^{2(a),(c)}$, shown in Figure 6-20. Overall, the vertical modal response of the decoupled model is not significantly different from the

NPM wet model without Belleville washers ($\{\{ \}^{2(a),(c)}$ vertical mass participation in the $\{\{ \}^{2(a),(c)}$ frequency range), that was used for the generation of the simplified beam model in Section 6.4.

In addition to aiding the creation of a simplified model of the NPM, the detailed 3D finite element model provides the initial finite element model that is later expanded to include the entire RP (see Section 3.1.5).

3.1.2 Creation of a Simplified Model of the NuScale Power Module in ANSYS and SAP2000

This step creates a linear beam model of the NPM in ANSYS and SAP2000 that have dynamically equivalent behaviors as the 3D model, but have fewer elements and nodes. This step does not perform the seismic analysis of the NPM, but develops a model for future analysis. Multiple analyses including modal, harmonic, and time history transient analyses are performed to tune the models to match the dynamic response of the detailed 3D model. Tuning is an iterative process in which analysis is performed on the simplified model and the results are compared to those of the detailed 3D model. Modifications expected to improve the comparison are then made to the simplified model and the analyses and comparisons are repeated. Modifications that are made to the model may include adjustments to the mass distribution and stiffnesses. This process is repeated until the results of the simplified model replicate those of the detailed 3D model. The simplified model of the NPM is further described in Section 6.0.

3.1.3 Creation of the Reactor Building Model in SAP2000

This step develops a detailed finite element structural analysis model of the RXB using the SAP2000 computer program. The simplified NPM SAP2000 model is inserted into each of the twelve reactor bays in the RXB model. This model is converted to the SASSI RXB model for the SSI analyses in a later step. The details and methodology used for the RXB SAP2000 model are discussed in the NuScale FSAR Section 3.8.4.

3.1.4 Creation and Analysis of a Reactor Building Model in SASSI

The SAP2000 RXB model created in step 3.1.3 is converted to a SASSI model. The SASSI RXB model is then evaluated for SSI by analysis performed in the frequency domain. The results of this analysis are used for seismic analysis of the RXB; however, because only a simplified representation of the NPM is included, final seismic analysis of the NPM is not performed with this model. Results from this analysis are used only as inputs for seismic analysis of the detailed 3D model of the NPM, as described in a later step. Results from the RXB seismic analysis include in-structure acceleration time histories at each NPM support, at centerline nodes of the CNV, and various locations on the RP walls and floor. In-structure response spectra are also calculated. The RXB analysis is repeated for multiple SSI analysis cases. The details and methodology used for the RXB SASSI model are discussed in the NuScale FSAR Section 3.7.2.

3.1.5 Creation and Analysis of Detailed Three-Dimensional NuScale Power Module Models in ANSYS (Entire Pool Model)

Detailed 3D finite element models of the NPM and RP fluid are created and used to perform seismic analysis. The 3D model created in the initial step of the seismic design process (see Section 3.1.1) is used and modified to include the entire RP volume (i.e., “entire pool model”). Multiple versions of the model are created to analyze the bounding NPM locations within the RXB. The detailed 3D models of the NPM are further described in Sections 4.0 and 5.0.

Seismic analyses of the models are performed by applying time history displacements from the Reactor Building SSI analysis to the NPM support locations, and time history accelerations to the RP fluid surfaces in contact with the RP floor and walls.

At acoustic fluid element surfaces, possible boundary conditions for transient dynamic analysis include either specified pressure or specified normal acceleration time histories. For the acoustic element surfaces, specified displacement is not an option provided by ANSYS. Acceleration time histories are applied on fluid boundaries at the reactor pool walls and floor and at fluid surfaces where NPM are not explicitly modeled. Zero pressure is specified on the top surface of the pool.

Displacement time histories are applied to structural supports where displacement degrees of freedom exist (NPM skirt support and lugs). Displacement time histories used for the structural supports are obtained by double integration of the acceleration time histories.

When acceleration time histories taken from SASSI are double integrated and input to the ANSYS analysis, drift or “baseline errors” may be introduced due to choice of the integration methods and initial conditions. The drift is eliminated by introducing a small adjustment to the accelerations at the beginning of each record. The drift correction has negligible effect on the maximum response occurring during the strong motion portion of the excitation.

The use of enforced accelerations represents an alternate means to prescribe boundary conditions at the structural supports. The integration time step used in the analysis is sufficiently small that the two approaches produce approximately the same results.

In the detailed 3D models, only one of the twelve NPMs is modeled. To account for the effect of NPMs that are not explicitly modeled, CNV centerline time history accelerations are applied to the surfaces of the fluid that would be in contact with the “missing” NPMs.

Multiple seismic analyses, described in Section 8.0, are performed for each model. Results generated during the analysis include maximum reaction and internal forces and relative displacements at various locations within the NPM. The NPM model also stores results for the creation of ISRS and time histories (displacement and reaction force).

3.1.6 Generation of In-Structure Time Histories and Response Spectra

Further processing of the results obtained from the seismic analyses of the detailed 3D NPM models is performed in order to produce inputs for downstream analyses. At various in-structure locations, one vertical and two horizontal sets of time history displacements and accelerations are generated using the seismic analysis results from the detailed 3D NPM model, each set representing the effect of three components of statistically independent earthquake motions applied simultaneously.

ISRS are generated by post-processing the analysis results in ANSYS. ISRS are developed in accordance with Regulatory Guide (RG) 1.122 (Reference 10.1.6). Two horizontal and a vertical response spectra are computed from the time history motions of the supporting structure at elevations of interest. Because the mathematical model of the supporting structural system (i.e., RXB) is subjected simultaneously to the action of three statistically independent spatial components of an earthquake, the three computed in-structure time histories used to compute the three ISRS account for all components of seismic input ground motion. Design spectra are generated by broadening and enveloping ISRS computed for applicable soil and rock profiles and concrete conditions. Design response spectra are provided for 2, 3, 4, 5, 7 and 10 percent damping.

3.1.7 Seismic Analysis of NuScale Power Module Components

The final step in the seismic design of the NPM is to perform stress analysis of the NPM components using inputs developed in previous steps. As appropriate to each component, seismic analysis methods and procedures satisfy relevant sections of American Society of Civil Engineers (ASCE) 4-98 (Reference 10.1.7), ASCE 43-05 (Reference 10.1.8), ASME Boiler and Pressure Vessel Code (Reference 10.1.5) or IEEE-344-2004 (Reference 10.1.4). ASCE4-98 is in the process of being updated to ASCE 4-13 (Reference 10.1.1) and is also referenced for guidance. Additional guidance for analysis of the NPM is provided in Section 7.0.

4.0 Detailed Three-Dimensional ANSYS models of the NuScale Power Module and Single Bay Pool

This section describes 3D finite element models of the NuScale Power Module (NPM) that are used for seismic analysis. These models use time history motions from the reactor building (RXB) analysis as inputs to the NPM support locations and pool walls and floor. The NPM models store results for creation of ISRS, time histories (displacement and acceleration), and maximum reaction and internal forces at various locations within the NPM. This information allows for analysis of the CNV, the RPV, the RVIs, the valves, the SGs, CRDMs, fuel, and other components within the NPM.

Three 3D ANSYS finite element models of the NPM are developed, which include the following variants:

- NPM with single bay pool
- NPM at location one with entire pool
- NPM at location six with entire pool

The first of these three variants is described in Section 4.1 and consists of submodels for a single CNV and nearby pool water within the reactor bay, the RPV, upper RVI, lower RVI, and CRDMs. The other two variants are described in Section 5.0 and consist of the same submodels as the first variant but with the pool water extended to the entire pool. The NPMs in operating bays 1 and 6 are representative of the other NPMs since the forces at the CNV support skirt and lug supports in bays 1 and 6 bound those of the other NPM locations. Therefore, the 3D models including the entire pool are developed for the NPM in operating bay 1 and in operating bay 6 in order to bound the response of all NPMs.

4.1 Seismic Model Methodology

The NPM is modeled primarily using solid elements, solid shell elements, beam elements, and acoustic fluid elements (for the RP). Pipe elements, matrix elements, surface elements, mass elements, contact/target elements, and spring elements are also used. A full 360-degree model is used because this model does not have symmetric boundary conditions that would justify using a partial symmetric model (half-model or quarter-model).

The NPM seismic model as shown in Figure 4-1 is created from five submodels: CNV and NPM pool bay, RPV, lower RVI, upper RVI, and CRDMs. ~~Four of these submodels (not including the CRDMs submodel) were created in ANSYS. All~~ The five submodels are combined into a single model. The details of the submodels and the combined model are described in the following sections. The coordinate system of the models is shown in ~~in~~ Figure 4-2.

The term “submodel” does not refer to the term “substructure” as used by the ANSYS structural analysis program. The term “submodel” refers to assemblies or components that are separately defined in the ANSYS input. To form a single model, the submodels

are connected by constraint equations, contact elements, or coupled degrees of freedom. ANSYS also uses the term “submodeling” to describe a finite element technique that can be used to obtain more accurate results in a particular region of a model, by using a more refined model of the particular region. This use of the term “submodel” is not the meaning intended in this report. In the ANSYS documentation, substructuring refers to procedures that condense a group of finite elements into one element represented by a single mass, stiffness and damping matrix. The single-matrix element is called a “superelement.” The ANSYS substructuring technique, ANSYS superelements, and the concept of master nodes are not used in this report.

The CRDM support frame and CRDM submodel described in this report are generated by the ANSYS computer code. The ANSYS CRDM submodel is translated from a CRDM stress analysis model developed by the CRDM vendor using their proprietary structural computer code. Modal analyses were performed to verify the equivalence of the ANSYS to the CRDM vendor models.

The piping, valves, manways, instruments, PZR heaters, and other small internal components such as bolts are not explicitly modeled. These minor features do not affect the gross structural behavior of the model and removing them allows for simplified meshing techniques to be used. The piping is flexible relative to the vessels, so it does not drive the response of the CNV or RPV.

Potential uplift of the NPM is captured through nonlinear contact with the rigid floor surface.

The three concrete pool walls and pool floor are modeled by applying no boundary conditions to these surfaces. The default boundary condition for the external faces of the acoustic body is to reflect acoustic pressure waves, which is the desired behavior for the concrete walls and floor.

The side of the NPM bay that is open to the rest of the pool is modeled as a reflective rigid wall. The pressure reflection at this interface is explicitly modeled with the rest of the pool, as described in Section 5.0. The entire pool model is used for transient analysis, described in Section 5.1.

The top surface of the pool is assigned an acoustic pressure of zero psi because this is the free surface.

4.1.1.3 Containment Vessel and Pool Materials

The CNV is assigned material properties of SA-508 Grade 3 Class 2 steel for the upper CNV and SA-182 FXM-19 steel for the lower CNV. The elastic moduli are taken at the CNV operating temperature of 100 degrees F. The densities are taken at the as-built temperature of 70 degrees F. The material property values are derived from the 2013 ASME Boiler and Pressure Vessel Code, Section II.

The TAMAS is assigned the elastic modulus of low carbon steel. The elastic modulus is taken at the CNV operating temperature of 100 degrees F. The initial density of this material is arbitrary since it is adjusted as part of the mass correction.

~~The pool is assigned acoustic material properties, which include the mass density (0.035876 lbm/in³) and speed of sound in water (60,042 in/sec) at 100 degrees F.~~The assigned material properties of the acoustic fluid elements used to represent pool water at 100 degrees F are shown in Table 4-4:

Table 4-4 ANSYS Material Properties for Acoustic Fluid Elements Assigned to the Pool Water

<u>Property</u>	<u>ANSYS Material Property Label</u>	<u>Value</u>	<u>Units</u>
<u>Mass density</u>	<u>DENS</u>	<u>ρ</u>	<u>$(\text{lbf} \cdot \text{sec}^2)/\text{in}^4$ (lbm/in^3)</u>
<u>Dynamic viscosity</u>	<u>VISC</u>		<u>$(\text{lbf} \cdot \text{sec})/\text{in}^2$</u>
<u>Sonic velocity</u>	<u>SONC</u>		<u>in/sec</u>
<u>Bulk viscosity</u>	<u>BVIS</u>	<u>$\gamma^{2(a),(c)}$</u>	<u>$(\text{lbf} \cdot \text{sec})/\text{in}^2$</u>
<u>Thermal conductivity</u>	<u>KXX</u>	<u>Not applicable to structural analysis; zero value is assigned</u>	
<u>Specific heat</u>	<u>C</u>	<u>Not applicable to structural analysis; zero value is assigned</u>	
<u>Heat coefficient</u>	<u>CVH</u>	<u>Not applicable to structural analysis; zero value is assigned</u>	

4.1.2 Reactor Pressure Vessel Submodel

4.1.2.1 Reactor Pressure Vessel Geometry, Mesh and Mass

The RPV geometry is based on the RPV drawings. The computer-aided design model used to generate the drawings was defeatured and simplified in order to reduce the element count of the mesh. Figure 4-4 shows the simplified RPV geometry.

The steam generator assembly is comprised of two interwoven helical steam generators that feed two individual steam lines. The support structures are shared between each steam generator.

The steam generator assembly spans the annulus between the reactor pressure vessel and the upper riser. The steam generator tubes are supported by 8 locations of 21 tube support bar assemblies. The tube support bar assemblies are long with a relatively small cross section. These tube support bar assemblies hang from the T-shaped upper tube support bars on the integral steam plenum plate. The bottom ends of the tube support bar assemblies are supported by the lower tube support cantilevers. The lower tube supports only provide circumferential restraint for large displacements of the tube support bars and do not provide restraint in other directions.

The steam generator tubes rest on the tube support bar assemblies, providing radial and vertical support but allowing the tubes to slide axially along the tube length. The tubes are constrained at the feed and steam plenum tubesheets. The tube support bar assemblies are interconnected in the circumferential direction via the overhanging tabs from adjacent supports. This coupling effectively ties the tube support assemblies together in the circumferential direction.

The tube support bar assemblies also interface with each other radially outward between the upper riser and the RPV shell. This allows loads to be transmitted radially outward from the riser to the shell. The upper riser is coupled to the RPV shell to model this behavior. See Section 4.1.4.2 for further details on the upper riser to reactor vessel coupling. The stiffness of the steam generator assembly in the other directions is inherently flexible compared to the RPV and therefore, not modeled. The comparative stiffness of the reactor pressure vessel, including the steam generator properties beyond its mass contribution, and the radial coupling of the upper riser to RPV shell, do not have a significant effect on the results presented in this report.

The aspects of the steam generator assembly that are physically modeled are the lower tube support cantilevers and upper tube support bars on the integral steam plenum plate. The lower tube support cantilevers are attached to the reactor pressure vessel. The tip of the lower tube support cantilever is near, but not welded or bolted to, the upper riser lateral support pad (located on the upper riser at the elevation of the support cantilever). In the NPM model, coupling of the pilot nodes of the cantilever tip and support pad is assigned using constraint equations in the radial direction, shown in Figure 4-5. Each pair of pilot nodes was created at an identical location. The upper tube support bars are modeled as rectangular beams, integrally meshed directly to the integral steam plenum plate.

The RPV is meshed using 8-node solid shell elements and 8-node solid elements. The solid shell elements are used at any shell section where there is one element through the thickness. The solid elements are used at intersection regions of shells and where there is more than one element through the thickness. The RPV mesh is shown in Figure 4-4.

Figure 4-5 Connection between URVI and lower tube support cantilever SG supports RPV submodel

The RPV has its mass adjusted in a similar manner as the CNV. The RPV mass adjustment summary is shown in Table 4-4. A depiction of the RPV sections is shown in Figure 4-9. The cable mass in the vicinity of the CRDM supports is accounted for by increasing the density of the CRDM support structure. See Table 4-5 for the density adjustment calculation.

4.1.2.2 Reactor Pressure Vessel Boundary Conditions

~~The male end of the lower lateral RPV support is coupled to the female end on the CNV in the horizontal directions only. This is done using the pilot nodes for the two mating surfaces of the RPV/CNV alignment feature.~~

~~The RPV upper support segments are connected to the CNV ledges in the vertical and circumferential directions only. This is done using pilot nodes scoped to the slots of the RPV support and holes of the CNV ledges.~~ Pilot nodes are created for both the CNV and the RPV, and coupled to represent the boundary conditions of the RPV. Pilot nodes are manually created within ANSYS Mechanical as remote points. Faces of geometry are selected and the remote point is scoped to the faces. Scoping refers to the geometry over which a boundary condition is applied. ANSYS uses multipoint constraint equations to generate a connection between the remote point and the meshed face to which the remote point is scoped. Those multipoint constraint equations comprise the function and input for the pilot nodes.

The remote points connecting the CNV and RPV submodels have options that specify them as deformable, meaning that the geometry to which each remote point is scoped is free to deform. A TARGE170 element is used at the remote point location, and CONTA174 elements at the scoped faces. For the TARGE170 element, all six degrees of freedom (DOF) are used in the multipoint constraint. The CONTA174 elements are constrained in the translational DOFs only. This deformable option is chosen because if the boundary conditions on the RPV were specified as rigid, the combined model would be constrained unrealistically. The remote points considered here are used to transmit loads between each submodel, and are abstractions that do not add inaccurate stiffness to the submodels.

At the bottom of the RPV, the pilot node associated with the mating surface of the male RPV alignment feature, and the pilot node associated with the mating surface of the female CNV alignment feature, are constrained to displace equally in the horizontal directions using the CE command in APDL. This is because the nominal design is a $\frac{1}{2}$ inch gap between the RPV and CNV alignment features in the cold condition. Modeling this horizontal gap as closed is a reasonable assumption. These nodes are not coupled in the vertical direction because the CNV alignment feature is designed to allow thermal expansion of the RPV alignment feature in the vertical direction without restraining motion.

The boundary conditions at the RPV Supports are governed by the nature of the connection with the CNV Ledges. The bolts which connect the RPV support slots to the CNV ledges are tensioned to accommodate thermal expansion of the RPV in the radial direction. Cylindrical coordinate systems are employed in order to capture this behavior using the CE command. Constraint equations are applied between the pilot nodes of the RPV supports and the CNV ledges in only the circumferential and vertical direction, because the rotation of the RPV about its vertical axis is not possible due to the four bolted connections securing it to the CNV ledges. By coupling the displacement of the pilot nodes of the RPV supports to the displacement of the pilot nodes of the CNV

ledges in the circumferential direction, the connection is effectively modeled. Likewise, the bolted connection at each of the four CNV ledges generates the need to couple the pilot nodes associated with the slots of the RPV supports and the holes of the CNV ledges in the vertical direction (i.e., there is no uplift).

The CRDM support frame is modeled with beam elements as shown in Figure 4-20. The cross-sections of these beam elements are either a rectangle (representation of the support plates) or hollow rectangles (other beams in the CRDM support frame). An actual representation of the seismic support plates in Figure C-8 is not needed at the level of detail of the NPM seismic model. The connection to the RPV head is facilitated by six degree of freedom (DOF) target/contact pairs (bonded) to the proximal surfaces of the RPV head solid elements.

4.1.2.3 Reactor Pressure Vessel Materials

The RPV is assigned material properties of SA-508 Grade 3 Class 2 steel, except for the upper RPV support, which is SA-533 Grade B Class 2. The elastic modulus values are taken at the average reactor coolant system (RCS) temperature of 550 degrees F. The density is taken at the as-built temperature of 70 degrees F because the model is built with room temperature nominal dimensions.

4.1.3 Lower Reactor Vessel Internals Submodel

4.1.3.1 Lower Reactor Vessel Internals Geometry, Mesh and Mass

The lower RVI geometry is based on the lower riser and core support drawings. The computer-aided design model used to generate the drawings was defeatured and simplified in order to reduce the element count of the mesh. Figure 4-6 shows the simplified lower RVI geometry.

The lower RVI is meshed using 8-node solid shell elements and 8-node solid elements. The solid shell elements are used at any shell section where there is one element through the thickness. The solid elements are used at intersection regions of shells and where there is more than one element through the thickness. The reflector is modeled as a separate part as is the rest of the lower RVI. This avoids requiring a conformal mesh between the two parts. A cutaway view of the lower RVI mesh is shown in Figure 4-6 (fuel assembly beam elements not shown).

{

}}^{2(a),(c)}

Figure 4-6 Lower reactor vessel internals geometry and mesh

The lower RVI has its mass adjusted in a similar manner as the vessels. The lower RVI mass adjustment summary is shown in Table 4-7. The core support mass is the sum of the following three RVI components: the Core Support, Surveillance Capsules, and Core Entrance Flow Plate. The negative mass adjustment associated with the core support was incorporated by reducing the density of the reflector material as shown in Table 4-8. The meshed mass is higher than the actual mass because the cooling channels in the reflector are not modeled (they are filled in) in the ANSYS model.

4.1.3.2 Fuel Modeling

Within the lower RVI submodel is a model to represent the fuel assemblies. The model uses properties provided directly by the fuel vendor. The fuel model consists of beam and spring elements, as shown in Figure 4-8. The spring element only includes a rotational DOF. The diagram of a single fuel assembly beam model is shown in Figure 4-9. To account for all 37 fuel assemblies, the stiffnesses of the beams and springs were multiplied by 37, ~~and t.~~ The final values are listed in Table 4-9. Increasing the axial area of the beam elements by 37 also increases the mass by 37 for a total fuel mass of 30,710 lbm. The total mass does not include the fluid mass. The fluid mass determined in Table 4-14 is added separately to the NPM full model. The spring stiffness values are not calculated. The values are tuned based on a benchmark of the fuel beam model to experimentally measured frequency values for a single fuel assembly. The combined mass and stiffness for a single fuel assembly is attached to the upper and lower core plates of the lower RVI submodel (fixed at the base and laterally/fully rotationally coupled on the top of the fuel assembly, as shown in Figure 4-7.

Table 4-9 Fuel beam element properties

Combined Area, Stiffness & Elevations of Fuel Element	
Area, A [in ²]	{{ }} ^{2(a),(c)}
Moment of inertia, I _{xx} , I _{zz} [in ⁴]	{{ }} ^{2(a),(c)}
Polar moment of inertia, J [in ⁴]	{{ }} ^{2(a),(c)}
Thickness (square), T _{xx} , T _{zz} [in]	{{ }} ^{2(a),(c)}
BN ⁽¹⁾ upper surface elevation, y ₁ [in]	{{ }} ^{2(a),(c)}
Lower HMP ⁽²⁾ grid elevation, y ₂ [in]	{{ }} ^{2(a),(c)}
HTP ⁽³⁾ 1 grid elevation, y ₃ [in]	{{ }} ^{2(a),(c)}
HTP 2 grid elevation, y ₄ [in]	{{ }} ^{2(a),(c)}
HTP 3 grid elevation, y ₅ [in]	{{ }} ^{2(a),(c)}
HTP 4 grid elevation, y ₆ [in]	{{ }} ^{2(a),(c)}
TN ⁽⁴⁾ lower surface elevation, y ₇ [in]	{{ }} ^{2(a),(c)}
Lower nozzle spring stiffness, K _{θ,LEG} ⁽⁵⁾ [in-lbf]	{{ }} ^{2(a),(c)}
Fuel spring stiffness, K _{θ,ISG} ⁽⁶⁾ [in-lbf]	{{ }} ^{2(a),(c)}
Upper nozzle spring stiffness, K _{θ,UEG} ⁽⁷⁾ [in-lbf]	{{ }} ^{2(a),(c)}

Note:

(1) BN: Bottom Nozzle.

(2) HMP: HMP is a spacer grid product name.

(3) HTP1-4: HTP is a spacer grid product name. The numbers 1 through 4 represent the spacer grid location, with 1 being the lowermost location, 2 and 3 being intermediate locations, and 4 being the uppermost location.

(4) TN: Top Nozzle.

(5) LEG: Lower End Grid (HMP).

(6) ISG: Intermediate Spacer Grid (HTP).

(7) UEG: Upper End Grid (HTP).

}}

}}^{2(a),(c)}

Figure 4-7 Fuel core model (with real element shapes) and its connections to LRV

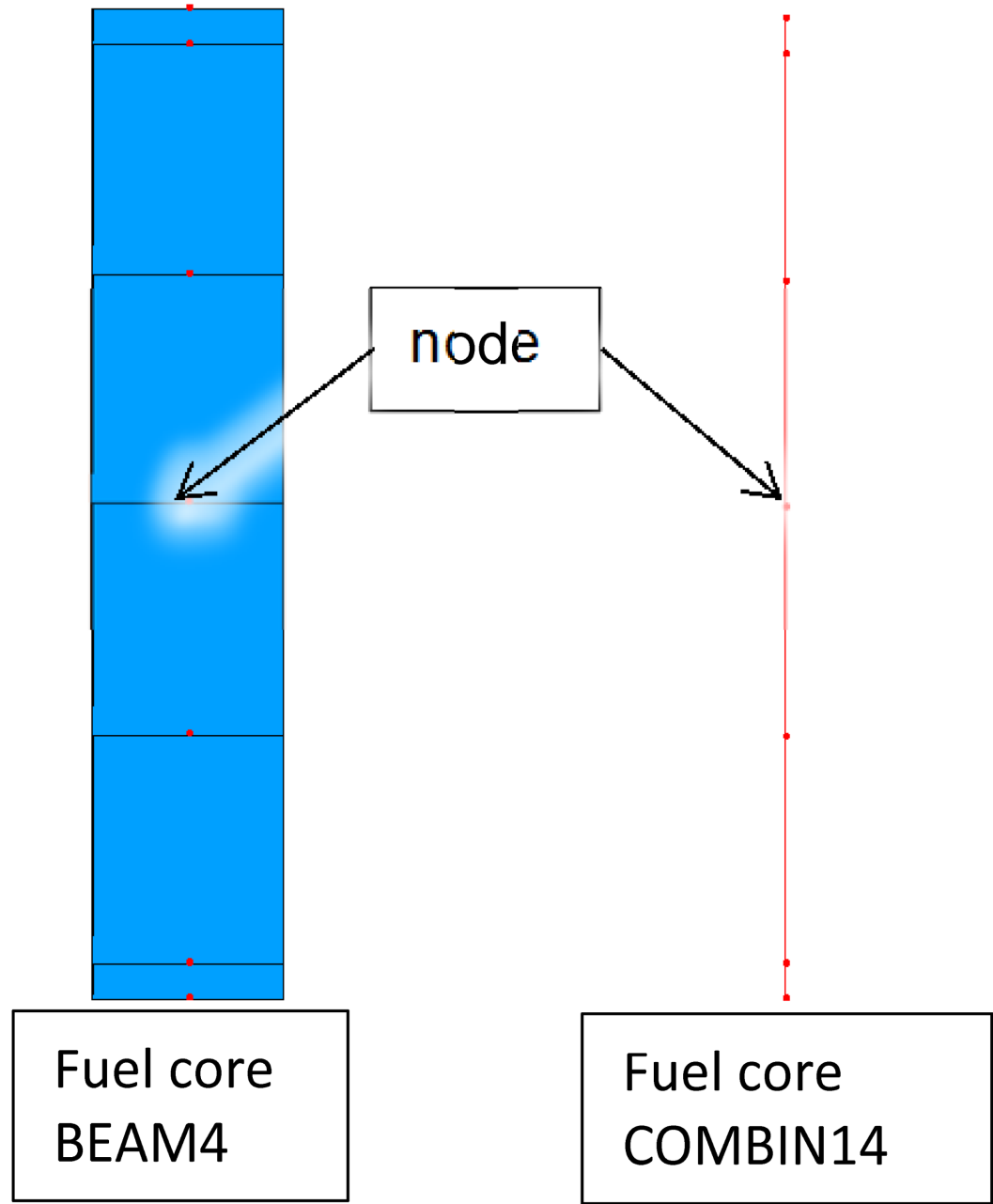


Figure 4-8 Fuel core model elements (real element shapes) and nodes

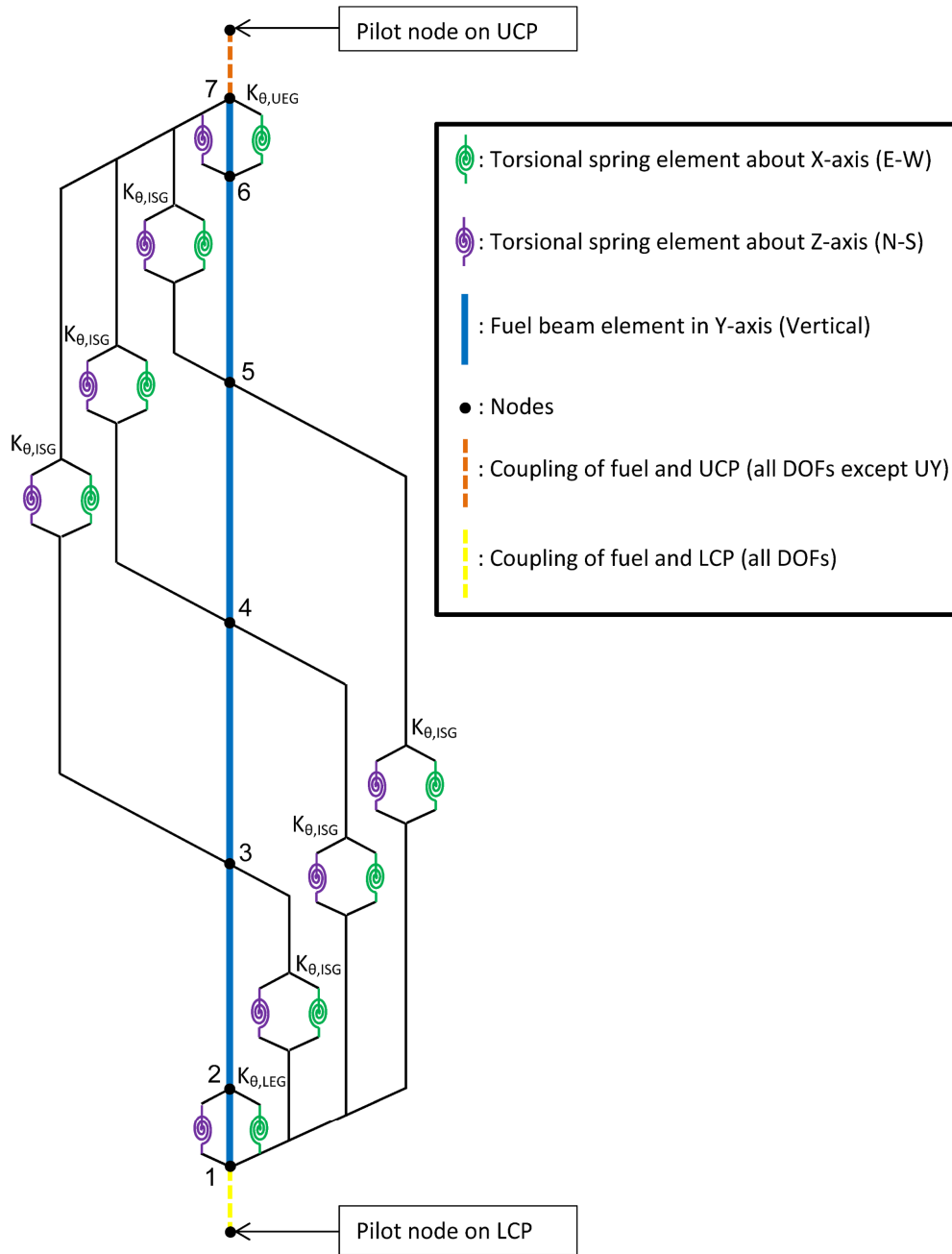


Figure 4-9 Fuel beam model

A modal analysis of the lower RVI was performed to verify the modes of the fuel assembly match the modes calculated by the fuel vendor. The entire lower RVI model except the fuel assembly was restrained for this analysis. The modal results are shown in Table 4-10.

Table 4-10 Fuel assembly modal results validation

NuScale Results		Fuel Vendor Results	
X-Frequency (Hz)	Z-Frequency (Hz)	X-Frequency (Hz)	Z-Frequency (Hz)
{{			}} ^{2(a),(c)}
{{			}} ^{2(a),(c)}
{{			}} ^{2(a),(c)}
{{			}} ^{2(a),(c)}
{{			}} ^{2(a),(c)}

4.1.3.3 Lower Reactor Vessel Internals Boundary Conditions

~~The four lower core plate tabs on the lower RVI are coupled to the four lower core support blocks of the RPV submodel in the horizontal direction. For the vertical boundary condition, four spring elements are added between the four lower core plate tabs and the four lower core support blocks. These springs represent the Belleville washers at this connection.~~ In separate submodels for the Lower RVI and the RPV, pilot nodes are created using the detailed methodology described in Section 4.1.2.2. The four lower core plate tabs on the lower RVI are coupled to the four lower core support blocks of the RPV submodel in the horizontal direction using these pilot nodes. For the vertical boundary condition, four COMBIN14 spring elements span the pilot nodes of the four lower core plate tabs and the pilot nodes of the four lower core support blocks.

The spring constant applied to the COMBIN14 elements at these locations was chosen to tune the core's vertical natural frequency away from the peak vertical acceleration frequencies observed in prior seismic analysis without the Belleville washers. To reduce the response of the core to the peak accelerations near {{ }}^{2(a),(c)}, a target natural frequency of {{ }}^{2(a),(c)} was selected for the core. To achieve this target vertical frequency, the combined spring constant of the 10 Belleville washers acting in series at each core support block was calculated to be {{ }}^{2(a),(c)}. Each washer was sized to avoid overstress, to avoid flattening out, and to keep the lower core plate tabs from touching the core support blocks during a seismic event. Additionally, since Belleville washers cannot carry loads in tension, three washers are included above each tab to prevent uplift and maintain the desired spring rate when the Belleville washers beneath each tab are no longer in compression. When acting in series, the three washers above each core plate tab have a combined spring constant of {{ }}^{2(a),(c)}. The three washers above each core plate tab are restrained vertically by the retaining nut above them.

Figure 4-10 shows the meshed finite element model with one element of the core support block made transparent for viewing the overlapping pilot nodes of the lower core plate tab and the core support block. These nodes overlap initially, but move relative to each other vertically once gravity and seismic loads are applied. The COMBIN14 element that models the Belleville washers spans these nodes.

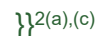


Figure 4-10 Meshed finite element model highlighting the COMBIN14 element that models the Belleville washers at this location. (The element representing the core support block is transparent for clarity.)

There are no nonlinear effects, such as gaps, in the connection between the Lower RVI core plate tabs and the core support blocks on the RPV because the Belleville washers are springs themselves, represented by linear spring elements.

The upper support blocks of the lower RVI are connected to the inner walls of the core region of the RPV submodel. This is done using a “no-separation” contact between the four pairs of mating surfaces. “No-separation contact” means contact detection points that are either initially inside the pinball region or that once they involve contact, always attach to the target surface along the normal direction to the contact surface (sliding is permitted). No friction coefficient is assigned to the elements. The no-separation contact is assigned using element types CONTA174 and TARGE170. These elements do not represent compression-only one-way springs.

Likewise, the reflector is connected to the inner surface of the core support barrel using a no-separation contact.

The no-separation contact surfaces are indicated in Figure 4-6. The contact surface meshes are shown in Figure 4-11 and Figure 4-12. Nonlinear effects are not considered for the radial gaps in the boundary conditions because the radial gaps are small (e.g. gap between reflector blocks and core barrel is 0.125 inch). Therefore, they are modeled as linear supports.

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11^{2(a),(c)}

Figure 4-11 Upper support blocks to RPV shell contact meshes

Figure 4-12 Lower RVI to Reflector contact meshes

4.1.3.4 Lower Reactor Vessel Internals Materials

The lower RVI is assigned material properties of Type 304 stainless steel. The elastic modulus is taken at the average RCS temperature of 550 degrees F. The density is taken at the as-built temperature of 70 degrees F.

4.1.4 Upper Reactor Vessel Internals Submodel

4.1.4.1 Upper Reactor Vessel Internals Geometry, Mesh and Mass

~~The upper RVI geometry is based on the upper riser drawings. The computer-aided design model used to generate the drawings was defeatured and simplified in order to reduce the element count of the mesh. Figure 4-7 shows the simplified upper RVI geometry.~~

~~The upper RVI is meshed using 8-node solid shell elements and 8-node solid elements. The solid shell elements are used at any shell section where there is one element through the thickness. The solid elements are used at intersection regions of shells and where there is more than one element through the thickness. A cutaway view of the upper RVI mesh is shown in Figure 4-7.~~ The upper RVI geometry is based on the upper riser drawings. The upper RVI assembly is primarily composed of the upper riser shell.

The upper riser shell is a long cylindrical structure that is approximately $\{ \}$ $\}^{2(a),(c)}$ The bottom of the upper riser shell is attached to a cone that connects the upper riser to the lower riser. The upper riser shell is bolted to the pressurizer baffle plate by the upper riser hanger ring. The upper riser hanger ring is a $\{ \}$ $\}^{2(a),(c)}$ plate. The upper riser hanger braces connect the upper riser shell to the upper riser hanger ring.

Within the upper riser shell are five control rod drive shaft supports. These supports guide the CRDM shafts from the CRDMs to the fuel and provide guidance and support for the in-core instrumentation. As a secondary role, these supports also stiffen the upper riser shell. The members of these supports are approximately $\{ \}$ $\}^{2(a),(c)}$ tall.

The upper riser bellows are between the upper riser shell and the upper riser cone section. The bellows allow for vertical thermal growth while limiting relative horizontal deflections between the upper riser and the lower riser. While the geometry of the bellows has not been modeled, its effect has been captured by coupling the upper riser and lower riser in the horizontal directions while the vertical directions are not coupled.

The in-core instrumentation guide tubes, riser backing strips, and the RCS injection piping from the CVCS are not modeled. None of these structures significantly affect the stiffness of the upper riser assembly and therefore do not affect the analysis results. The upper riser geometry and general dimensions are shown in Figure 4-13.

The CAD model used to generate the drawings was defeatured and simplified in order to reduce the element count of the mesh. Figure 4-14 shows the simplified upper RVI geometry.

The upper RVI submodel is composed of the upper riser shell, upper control rod drive shaft supports, upper riser hanger ring, and upper riser hanger braces. The upper riser shell and upper riser hanger ring are meshed with SOLSH190 elements while the upper control rod drive shaft supports and upper riser hanger braces are meshed with SOLID185 elements. Figure 4-14 presents the mesh of the upper riser. SOLID185 elements are shown in red and SOLSH190 elements are shown in blue. The upper RVI has its mass adjusted in a similar manner as the lower RVI. The upper RVI mass adjustment summary is shown in Table 4-11. See Section 4.1.8.2 for additional details on mass elements added to the model.

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11^{2(a),(c)}

Figure 4-13 Upper Riser Geometry

{{

}}^{2(a),(c)}

Figure 4-14 Upper reactor vessel internals geometry and mesh

~~The upper RVI has its mass adjusted in a similar manner as the lower RVI. The upper RVI mass adjustment summary is shown in Table 4-10.~~

Table 4-11 Mass adjustment summary for the upper RVI

Upper RVI Section	Mass (lbm)	Mesh Mass (lbm)	Mass Adjustment (lbm)
Upper riser (total)	{{		}} ^{2(a),(c)}
		TOTAL Upper RVI MASS (lbm):	{{ }} ^{2(a),(c)}

4.1.4.2 Upper Reactor Vessel Internals Boundary Conditions

The physical connection between the upper RVI and the lower RVI is shown in Figure C-18 and described in Table C-1, Interfacing component, "Upper riser with lower riser." The cone of the upper RVI is coupled to the cone of the lower RVI submodel in the horizontal directions only, as shown in Figure 4-15.

~~Eight~~ The physical connections between the upper riser and the RPV are shown in Figure 4-16 and Figure 4-17. In both the single bay model and the entire pool model, ~~eight~~ rectangular contact surfaces on the upper RVI are coupled to the tips of the lower radial cantilever SG supports RPV submodel by coupling the eight pairs of pilot nodes in the radial direction only. Each pair of pilot nodes was created at the same location to avoid an inaccurate constraint. ~~Radial~~ In the entire pool model, radial coupling is provided between the upper riser and the RPV using constraint equations, as shown in Figure 4-18. These represent the radial load transfer that occurs due to the stack-up of SG tube supports between the upper riser and RPV. The load transfer occurs along the height of the SG at the 8 support locations around the circumference.

Note the Single Bay model, used for the generation of the simplified beam model shown in Section 6.4, does not have radial coupling between the upper riser and the RPV. Overall, the horizontal harmonic response of the Single Bay models with and without radial coupling between the upper riser and the RPV is not significantly different. The harmonic force amplitudes at key locations from these two models are compared, and the results are comparable.

The connections between the upper RVI and the baffle plate are shown in Figures C-14 and Figure C-15, and described in Table C-1, Interfacing component, "Upper riser hanger ring with the PZR baffle plate." The upper riser ring hole locations on the upper RVI are coupled to pin locations on the baffle plate of the RPV submodel. This is done by coupling the translational degrees of freedom on the eight pairs of pilot nodes, as shown in Figure 4-19. See Section 5.0 for an additional constraint that applies only to the entire pool models.

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11^{2(a),(c)}

Figure 4-15 Constraint equations between URVI and LRVI

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11^{2(a),(c)}

Figure 4-16 Connections between upper RVI and RPV (top section view)

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11^{2(a),(c)}

Figure 4-17 Connections between upper RVI and RPV (side view)

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112(a),(c)

Figure 4-18 Constraint equations between the upper RVI and RPV (Entire Pool model)

Figure 4-19 Connection between URVI and baffle plate of RPV submodel

4.1.4.3 Upper Reactor Vessel Internals Materials

The upper RVI is assigned material properties of Type 304 stainless steel. The elastic modulus is taken at the average RCS temperature of 550 degrees F. The density is taken at the as-built temperature of 70 degrees F.

4.1.5 Control Rod Drive Mechanism Submodel

For the CRDM submodel, minor modifications were made to the mass and to couple the external coil stack beams to the pressure housing pipe elements. The CRDM submodel is imported 16 times into the combined model and connected to the RPV head, as shown in Figure 4-20.

The CRDM is meshed using 2-node pipe elements, 2-node beam elements and structural mass elements. Additional masses were added to account for the control rod drive shafts and control rod assemblies (CRAs). The vertical masses of the drive shafts

4.1.5.1 Control Rod Drive Mechanism Boundary Conditions

A CRDM is constrained to the RPV head by beam elements representing the CRDM nozzles. The nozzle elements share the two nodes through the thickness of the RPV head, and connect to the bottom of the CRDM by six DOF target/contact pairs (always bonded) to the proximal surfaces of the RPV head solid elements. This couples the translational degrees of freedom, as well as rotation about the two horizontal axes. CRDM rotation about the vertical axis is constrained.

The mid-heights of the CRDMs are coupled laterally to the CRDM support frame. Nonlinear effects are not included in the linear analysis. The radial gaps between the CRDM latch housings $\{\{ \quad \}^{2(a),(c)}$ and the seismic support plates $\{\{ \quad \}^{2(a),(c)}$ are sized nominally to provide a tight fit in the cold assembly condition.

The tops of the CRDMs are coupled laterally to the inside of the CNV top head opening, and the gaps are closed by the seismic support frame bolts.

4.1.5.2 Control Rod Drive Mechanism Support Structure Materials

The mid-height of the CRDM is coupled laterally to the CRDM support structure frame on the top of the RPV.

Material properties for the frame and support plates (connecting the frame to the CRDMs) are assigned as Type 304 stainless steel. The elastic modulus is taken at the average CNV gas temperature of 300 degrees F. The density is taken at the as-built temperature of 70 degrees F.

4.1.5.3 Control Rod Drive Mechanism Materials

The CRDM submodel uses several sets of material properties whose values were assigned at temperatures between 450 degrees F and 600 degrees F. An additional set of material properties for SA-508 Grade 3 Class 2 was added to represent the CRDM nozzle material at 550 degrees F.

4.1.6 Piping Fluid Mass Summary

The masses of the piping fluid applied to the previously described submodels are summarized in Table 4-12. The fluid masses apply to multiple submodels. Reactor component cooling water system fluid is neglected; however, as its volume is small, and its calculated mass is negligible. See Section 4.1.8.2 for computation of steam and feedwater (FW) densities. Feedwater density is also applied to chemical and volume control system piping fluid density. The decay heat removal system (DHRS) water density is taken at 60 degrees F and atmospheric pressure (62.4 lbm/ft³, or 0.0361 lbm/in³).

{

}}^{2(a),(c)}

Figure 4-21 Containment vessel and reactor pressure vessel section diagrams

4.1.8 Combined Model

The five submodels are imported into the combined NPM model. The combined model contains directional masses that are applied to the applicable surfaces of the NPM model. These masses include the RCS fluid mass, SG mass, control rod drive shaft ~~(CRDS)~~ mass, and CRA mass. Each has the horizontal and vertical masses applied to different surfaces. The total mass is divided by the number of nodes in the named selection, and a point mass is applied to each node in the named selection. The directional masses are explained in detail in Sections 4.1.8.1 through 4.1.8.4.

the bottom cantilever interface. See Figure 4-23 for the locations of the distributed mass. The SG ISRS presented in Appendix B envelops both steam generators. The actual configuration of the helical tube columns of the two SGs form an intertwined bundle of tubes around the upper riser assembly, with a total of four feed plena and four steam plena located 90 degrees apart around the RPV (shown in Figure 2-5). Therefore, the simplified SG modeling detail in the NPM seismic model is adequate for the overall NPM seismic response analysis. Subsequent component qualification remains unaffected.

The mass of the SG includes the tube mass, tube support mass, and secondary fluid mass (see Table 4-16). The mass of the cantilevers and upper support bars are included in the RPV mass adjustment (Section 4.1.2). The secondary fluid mass is calculated below.

The secondary fluid SG fluid mass is calculated in Table 4-15. The liquid and vapor temperatures are for 100 percent reactor power.

Table 4-15 Steam generator secondary fluid mass calculation

Section	Volume fraction of liquid and vapor	Temperature (°F)
Liquid	{{	}} ^{2(a),(c)}
Vapor	{{	}} ^{2(a),(c)}
Total volume (in ³)	{{ }} ^{2(a),(c)}	

Table 4-16 Steam generator mass summary

Component	Mass (lbm)
Tubes + supports	{{ }} ^{2(a),(c)}
Internal fluid	{{ }} ^{2(a),(c)}
TOTAL	{{ }} ^{2(a),(c)}

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112(a),(c)

Figure 4-23 Mass elements (red dots) representing SG mass

4.1.8.3 Control Rod Drive Shaft Mass

The CRDS control rod drive shaft horizontal masses are applied to the various lateral supports along the height of the shafts. The total CRDS control rod drive shaft mass is $\{\{ \}^{2(a),(c)}$ per shaft. The mass each support carries is $\{\{ \}^{2(a),(c)}$ times the ratio of the support span to total height. See Figure 4-11 for a diagram of the support elevations and spans. The vertical mass of the CRDS control rod drive shaft is applied in the CRDM submodel (see Section 4.1.5). The top two horizontal masses are also applied within the CRDM submodel. The remaining horizontal masses are applied in the combined NPM submodel.

$\{\{$

$\}^{2(a),(c)}$

Figure 4-24 Control rod drive system support names and span lengths

The radial gap between the control rod drive shaft alignment cone $\{\{ \}$ ^{2(a),(c)} and CRA hub $\{\{ \}$ ^{2(a),(c)} is closely controlled to provide a tight fit. The lowest CRA card is located only $\{\{ \}$ ^{2(a),(c)} above the top of the upper core plate (direct load path), and provides for an even closer fit to the control rods $\{\{ \}$ ^{2(a),(c)}. Therefore, gaps are not modeled.

4.1.8.4 Control Rod Assembly Mass

The mass of the control rod assemblies (CRA) is also applied directionally. Half of the horizontal mass is applied to the CRA guide tube support plate, and the other half is applied to the upper core plate. In the normal operating position, the CRA are all out of the core, and the rodlets contact the guide tube support cards. The hub on top of the CRAs contact the control rod drive shaft alignment cones, structurally connected to the top of the lower riser by the CRA guide tube support plate. The load path for lateral seismic loads is through the upper core plate on the bottom and to the support plate on top of the lower riser, that are included in the model (the guide tubes and support cards are not modeled).

In the vertical direction, the control rod drive shaft with attached CRA is supported during normal reactor operation (all rods out of the core) by the stationary grippers on top of the RPV (in the CRDM submodel), ignoring any friction on CRA and drive shaft supports. The vertical mass is combined with the **CRDS** control rod drive shaft mass in the CRDM submodel. Each CRA has a mass of $\{\{ \}$ ^{2(a),(c)}.

4.1.8.5 Fluid Coupling

The fluid coupling in the annular region between the RPV and the RVI is accounted for by a method called Fourier Nodes. This method accounts for the mass of the fluid as well as the fluid-structure interaction between the RPV and RVI. It reduces the computational effort without explicitly modeling acoustic elements (such as in the NPM pool bay). This method creates a set of constraint equations connecting the inner and outer surfaces of the annulus at several elevations along the annulus (23 locations for this model). as shown in Figure 4-25. At each elevation, two nodes are created at the center of the module as the Fourier nodes (identical location); one for RVI and the other for RPV. Each Fourier node is coupled to the wall inner surface in the radial direction at 16 locations along the circumference using the CE command.

The ANSYS element type MATRIX27 with symmetric element matrices (keyopt(2)=0) is used for fluid coupling. For each element, 78 matrix constants are entered to the mass matrix, as shown in Equation 4-1 below. In Equation 4-1, F_{xn}^a and F_{zn}^a are the reaction forces at Fourier Node a which is coupled to the inner cylinder in the x-axis and z-axis, and F_{xn}^b and F_{zn}^b are the reaction forces at Fourier Node b which is coupled to the outer cylinder in the x-axis and z-axis, respectively. For the beam mode, the hydrodynamic masses M_{11} , M_{12} , M_{21} , and M_{22} are expressed in Equation 4-2, which is from Reference 10.1.11

$$\begin{bmatrix} F_{xn}^a \\ F_{zn}^a \\ F_{xn}^b \\ F_{zn}^b \end{bmatrix} = \begin{bmatrix} M_{11} & 0 & 0 & 0 & 0 & 0 & M_{12} & 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ & M_{11} & 0 & 0 & 0 & 0 & 0 & 0 & M_{12} & 0 & 0 & 0 \\ & & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ & & & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ & & & & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ & & & & & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ M_{21} & & & & & M_{22} & 0 & 0 & 0 & 0 & 0 & 0 \\ & & & & & & 0 & 0 & 0 & 0 & 0 & 0 \\ & & & & & & & 0 & 0 & 0 & 0 & 0 \\ & & M_{21} & & & & & M_{22} & 0 & 0 & 0 & 0 \\ & & & & & & & & 0 & 0 & 0 & 0 \\ & & & & & & & & & 0 & 0 & 0 \\ & & & & & & & & & & 0 & 0 \\ & & & & & & & & & & & 0 \end{bmatrix} \begin{bmatrix} \ddot{u}_{xn}^a \\ \ddot{u}_{zn}^a \\ \ddot{u}_{xn}^b \\ \ddot{u}_{zn}^b \end{bmatrix} \quad \text{Equation 4-1}$$

$$M_{11} = \frac{\pi \rho L a^2 (1 + r^2)}{(1 - r^2)} \quad \text{Equation 4-2}$$

$$M_{12} = M_{21} = \frac{-2abr}{(1 - r^2)}$$

$$M_{22} = \frac{\pi \rho L b^2 (1 + r^2)}{(1 - r^2)}$$

where

M_{ii} = hydrodynamic mass ($i = 1$ or 2)

ρ = fluid density

L = axial length of the discretized segment

a = radius of the inner cylinder shell

b = radius of the outer cylinder shell

$r \equiv a/b$

Only the first coupled mode ($M=1$, beam mode) between the RPV and RVI is considered because shell modes do not have a significant impact on the overall response of the NPM. This is justified since the overall seismic inertial loading and response of the NPM is dominated by beam modes. Modal participation associated with local shell modes (determined without considering shell mode fluid coupling) is not significant. The frequency of shell modes such as the $n=8$ mode of the lower riser are above the range of significant seismic input and response is not sensitive to frequency changes due to fluid coupling. Therefore it was not necessary to include fluid coupling effects upon shell modes.

For calculation of the mass, the outer radius of the SG region of the annulus was reduced to adjust for the volume taken up by the SG (the last two rows of region 4, and

all of region 5, Table 4-17 and Figure 4-26). The adjusted volume is reduced by 34 percent, which is 1 minus the ratio of the RCS volume of the SG region $\{ \}$ $\}^{2(a),(c)}$ to the total volume of this region without consideration of the SG displacement $\{ \}$ $\}^{2(a),(c)}$.

Values for all five of these vectors are summarized in Table 4-17. Region location is shown in Figure 4-26. The geometric dimensions were obtained from the RPV and RVI geometry.

$\{ \}$

$\}^{2(a),(c)}$

Figure 4-25 Fourier node locations and couplings

$$f = \frac{1}{2\pi} \sqrt{\frac{k}{m}}$$

Equation 6-1

where,

f = frequency (Hz),

k = spring stiffness (lbf/in), and

m = effective mass of the 3D model with respect to each of the major modes (lbm).

For the wet condition, the effective mass used in Equation 6-1 is not derived from the 3D model, because the water mass is involved in the NPM response, and the water mass included at each major mode is unknown. Therefore, for the wet simplified beam model, harmonic analysis is used. The mass values for the wet simplified beam model are determined iteratively in the harmonic analysis to match the skirt vertical reaction force amplitudes from the 3D model results.

The masses and spring stiffnesses are listed in Table 6-2. For the dry model only three spring-mass combinations are used. For the wet model in Section 6.4, four spring-mass combinations are used.

Table 6-2 Vertical masses and spring stiffnesses

Spring stiffness number	Mass (lb·s ² /in)		Spring stiffness number	Spring-stiffness (lbf/in)	
	Dry	Wet		Dry	Wet
645	}}	}} ^{2(a),(c)}	615	}}	}} ^{2(a),(c)}
646	}}	}} ^{2(a),(c)}	616	}}	}} ^{2(a),(c)}
647	}}	}} ^{2(a),(c)}	617	}}	}} ^{2(a),(c)}
648	}}	}} ^{2(a),(c)}	618	}}	}} ^{2(a),(c)}

Mass Number	Mass (lb·s ² /in)		Spring Stiffness Number	Spring Stiffness (lb _f /in)		Frequency (Hz)
	Dry	Wet		Dry	Wet	
645	}}	}} ^{2(a),(c)}	615	}}	}} ^{2(a),(c)}	}}
646	}}	}} ^{2(a),(c)}	616	}}	}} ^{2(a),(c)}	}}
647	}}	}} ^{2(a),(c)}	617	}}	}} ^{2(a),(c)}	}}
648	}}	}} ^{2(a),(c)}	618	}}	}} ^{2(a),(c)}	}}

{{

}}^{2(a),(c)}

Figure 6-9 Reactor pressure vessel three-dimensional submodel for tuning and torsional mass moments of inertia

Table 6-3 Reactor pressure vessel horizontal masses and torsional mass moments of inertia

Node	Initial Mass (lbm)	Mass Tuning (lbm)	Final Mass (lbm)	Final Mass (lbm)	I _{zz} (lb·ft ² /in)	Additional Mass Sources
201	{{				}} ^{2(a),(c)}	
202	}}				}} ^{2(a),(c)}	
203	}}				}} ^{2(a),(c)}	
204	}}				}} ^{2(a),(c)}	
205	}}				}} ^{2(a),(c)}	
206	}}				}} ^{2(a),(c)}	
207	}}				}} ^{2(a),(c)}	
208	}}				}} ^{2(a),(c)}	
209	}}				}} ^{2(a),(c)}	
210	}}				}} ^{2(a),(c)}	
211	}}				}} ^{2(a),(c)}	
212	}}				}} ^{2(a),(c)}	
213	}}				}} ^{2(a),(c)}	
214	}}				}} ^{2(a),(c)}	
215	}}				}} ^{2(a),(c)}	
216	}}				}} ^{2(a),(c)}	
217	}}				}} ^{2(a),(c)}	
218	}}				}} ^{2(a),(c)}	
219	}}				}} ^{2(a),(c)}	
220	}}				}} ^{2(a),(c)}	
221	}}				}} ^{2(a),(c)}	
222	}}				}} ^{2(a),(c)}	
223	}}				}} ^{2(a),(c)}	
224	}}				}} ^{2(a),(c)}	
225	}}				}} ^{2(a),(c)}	}} ^{2(a),(c)}

As detailed in Tuning Section 6.2, the missing mass of the RVI (not captured in the spring-mass elements shown in Figure 6-11) is proportioned to the RPV upper and lower RVI support elevation nodes. Masses of RPV attachments that are not explicitly modeled in the underlying 3D vessel model (Figure 6-9), such as cables, are indicated for individual nodes and considered for their contribution to the torsional mass moment of inertia.

6.1.6 Reactor Pressure Vessel Supports

The RPV support skirts are modeled using four elements, as shown in Figure 6-10. An arbitrary small value is assigned as the cross section area of these four beams. High values of moment of inertia, shear modulus, and torsional constant are assigned to these four beams representing rigid elements in these directions.

At the end of each beam, three translational springs are applied. For these springs, one end is on the RPV support skirts (nodes 226 through 229) and the other end is on the CNV (nodes 128 through 131). The nodes in each spring are radially separated by 0.19 inches, determined from the 3D model to represent joint sliding in the radial direction. The RPV skirt joint is slotted to allow unrestrained thermal expansion between the CNV and RPV in the radial direction. For the other two directions (circumferential and vertical), the translational springs are connected to the RPV support ledge (on the CNV). Thus, these spring stiffnesses are calculated. Their stiffness is determined from static analysis on the 3D model.

Since in SAP2000 spring elements can only be defined in the global coordinate system (CS), the 45 degrees horizontal springs (local CS) are replaced by the springs in the global CS.

The dominant mode in the vertical direction is $\{\{ \quad \}\}^{2(a),(c)}$ for the dry model (See Table 6-7 and Figure 6-16) and $\{\{ \quad \}\}^{2(a),(c)}$ (See Figure 6-20) for the wet model. The dominant vertical modes consist of vertical translation of the RPV and RVI with relatively small translation of the CNV. The predominant vertical deflection occurs where the RPV upper support segments are connected to the CNV ledges. For the wet model, the fundamental mode at $\{\{ \quad \}\}^{2(a),(c)}$ (See Figure 6-19) is of lesser significance and consists of coupled vertical motion of the fluid, CNV, RPV and RVI.

Higher modes near $\{\{ \}^{2(a),(c)}$ (dry model shown on Figure 6-17 and Figure 6-18; wet model shown on Figure 6-21 and Figure 6-22) consist primarily of the vertical response of the top support structure and CNV.

 $\{\}$ $\} \}^{2(a),(c)}$

Figure 6-16 Dry model, 1st Significant Vertical Mode, $\{\{ \dots \} \}^{2(a),(c)}$

}}

}}^{2(a),(c)}

Figure 6-17 Dry model, 2nd Significant Vertical Mode, }}

}}^{2(a),(c)}

11

11^{2(a),(c)}

Figure 6-18 Dry model, 3rd Significant Vertical Mode, 11

11^{2(a),(c)}

11

11^{2(a),(c)}

Figure 6-19 Wet model, 1st Significant Vertical Mode, 11

11^{2(a),(c)}

{}

{}^{2(a),(c)}

Figure 6-20 Wet model, 2nd Significant Vertical Mode, {}

{}^{2(a),(c)}

11

11^{2(a),(c)}

Figure 6-21 Wet model, 3rd Significant Vertical Mode, 11

11^{2(a),(c)}

11

11^{2(a),(c)}

Figure 6-22 Wet model, 4th Significant Vertical Mode, 11

11^{2(a),(c)}

7.0 Seismic Analysis Methods for Structures, Systems, and Components that Comprise the NuScale Power Module

The NPM 3D model described in Section 5.0 was used to determine seismic inputs for the SSC that are integral to or attached to the NPM. Structures, systems, and components supported by the NPM can be analyzed by any of the dynamic analysis methods from NuScale FSAR Section 3.7.

7.1 Time History Analysis Method

For analysis of complex Structures, systems, and components within the NPM a more detailed structural model can be used with in-structure time histories obtained from the NPM 3D analyses. Qualification of fuel assemblies uses this approach.

7.2 Response Spectrum Analysis Method

The response spectrum method can be used for design of the SSC that are supported by the NPM where appropriate in accordance with Standard Review Plan (SRP) 3.7.2, SRP 3.7.3 and guidelines in ASCE 4.

From time history analyses of the NPM 3D model, time histories at locations of equipment supports within the NPM were calculated. Response spectra from the SSI cases were then enveloped and broadened, according to ASCE 4-13 (Reference 10.1.1) to give ISRS for use in design of the SSC supported within or directly on the NPM. The peak broadening method of ASCE 4-13 Section 6.2.3 (b), steps 1 to 5 was applied to the SSI cases. The spectra due to each SSI case was clipped and broadened before enveloping the SSI cases. Alternatively, the peak shifting method of ASCE 4-13 Section 6.2.3 (c) can be applied independently to each SSI case. The 15 percent reduction of narrow response peaks as defined in Section 6.2.3 (b) of ASCE 4-13 is not permissible in conjunction with the peak shifting method of Section 6.2.3 (c).

The in-structure floor response spectra provided in this report were generated using the guidance provided in RG 1.122, Rev. 1 (Reference 10.1.6). Reduction of narrow frequency peak amplitudes was not performed.

Analysis of piping supported by the NPM at multiple locations can be performed using the Uniform Support Motion (USM) approach. However, the USM method can result in considerable overestimation of seismic responses. Therefore, an alternate method that may be used is the independent support motion (ISM) method. The ISM method is generally used for piping systems that are supported by more than one structure, but may be used for piping systems with multiple supports located in a single structure, if appropriate. ~~The Independent Support Motion approach is not applicable to piping analysis.~~

7.3 Equivalent Static Load Method

Where applicable, the equivalent static load method can be used for equipment supported on or within the NPM. The input is the ISRS at the support point and the

8.0 Three-Dimensional Seismic Model Analysis

This section analyzes the NPM for seismic loading using the non-linear 3D ANSYS finite element models. Seismic time history data from the RXB are used as inputs to the model. Outputs of the model include ISRS, time history data, relative displacements, and forces and moments within the NPM.

The results were obtained using the entire pool seismic models described in Section 5.0. Results are included for both 'NPM 1 in the entire pool,' and 'NPM 6 in the entire pool' models.

The scope of this calculation includes analyzing the NPM for various seismic inputs, and generating time histories, relative displacements, ISRS, and forces and moments. The inputs include seismic time history data from the RXB seismic analysis for certified seismic design response spectra (CSDRS) input.

The NPM model is analyzed for the following:

- one seed input location (Capitola)
- one soil type (Soil 7)
- two RXB concrete conditions (cracked and uncracked)
- two modules (NPM 1 and NPM 6)
- one case nominal NPM stiffness for the uncracked case, and two NPM stiffness conditions for the cracked case ([1] NPM stiffness adjustment = $1/1.3=77\%$ of nominal stiffness and [2] nominal stiffness; i.e. no adjustment to NPM stiffness)

This gives 6 runs in total for the NPM models. The six runs are:

1. NPM 1 and entire pool, Cracked Concrete Properties, Nominal NPM stiffness
2. NPM 1 and entire pool, Cracked Concrete Properties, 77% of Nominal NPM stiffness
3. NPM 1 and entire pool, Uncracked Concrete Properties, Nominal NPM stiffness
4. NPM 6 and entire pool, Cracked Concrete Properties, Nominal NPM stiffness
5. NPM 6 and entire pool, Cracked Concrete Properties, 77% of Nominal NPM stiffness
6. NPM 6 and entire pool, Uncracked Concrete Properties, Nominal NPM stiffness

The analysis cases were performed using inputs derived from the SASSI analysis of the RXB with soil profile 7 (Hard rock) and a CSDRS compatible control motion based on the Capitola recording.

Outputs from the post-processing include:

- time-history displacement and acceleration data for 33 points within the NPM
- broadened ISRS for the same 33 points within the NPM
- maximum forces and moments at 77 interfaces between NPM components

- maximum forces and moments within 22 NPM component sections
- maximum forces at 4 NPM support locations

For each location, direction, and damping value, the response spectra were calculated for the six seismic analysis runs. Using each set of six response spectra, an envelope spectrum was constructed by finding the maximum of the six response values at each spectral frequency point. The envelope of the six spectra was then broadened by $\pm 15\%$ to produce the design ISRS (see example Figure 8-9 for the 4% damping curve in Figure B-11).

8.1 Transient Analysis

The input file (and subsequent APDL files that it executes) are set up to run CSDRS inputs.

The input files load the combined model as explained previously. The file has an option to reduce or increase stiffness of the NPM material properties and springs when not running the nominal stiffness cases. The commands in the file create a rigid floor and apply contact between the floor pilot node and the CNV skirt pilot node. This captures any uplift of the NPM.

The file then sets up the transient solution options, applies the table loads, and solves. Alpha-beta damping is used for the analysis, and a 7 percent damping value is applied to the frequency range of $\{\{ \dots \}^{2(a),(c)}\}$. This frequency range covers the major modes in Table 8-2, and the range is adjusted for the soft and stiff models. The use of 7 percent damping is justified in Appendix C.

A static time step with no loads is applied to ensure the model is in equilibrium before the transient starts. The initial time step for the static step is 1,000 seconds. A fixed time step of 0.001 seconds is used during the remaining transient portion of the solution.

For CSDRS runs, the transient run time is truncated to 24 seconds, plus the additional 1,000 second initial step, for a total of 1,024 seconds. This is acceptable since the main earthquake response for all provided seeds occurs in the first 24 seconds.

8.2 Modal Analysis

Modal analyses were run with and without the pool water (“wet” or “dry,” respectively). The Block Lanczos solver was used for the dry analysis, and the unsymmetric matrix solver used for the wet analysis. The total mass (static) of the model is 6942 lbf-s²/in, of which 2700 lbf-s²/in is the pool bay water mass. The sum of the percentage of modal mass to total mass in the horizontal directions exceeds 100% because of the effect of the confined hydrodynamic mass (see Section 6.6.5.2). Note that the bending modes of the major vessels include shell deformation.

Dry model modal results above a mass participation cutoff of 0.1% are shown in Table 8-1 and wet model modal results are shown in Table 8-2.

Table 8-1 Modal analysis results for the single-bay dry NPM model (no pool water)

ANSYS DRY MODEL	
X-Freq. (Hz)	X-Eff. Mass (lb·s ² /in)
1	
	2(a),(c)

ANSYS DRY MODEL	
Z-Freq. (Hz)	Z-Eff. Mass (lb·s ² /in)
1	
	2(a),(c)

ANSYS DRY MODEL	
Y-Freq. (Hz)	Y-Eff. Mass (lb·s ² /in)
1	
	2(a),(c)

}}2(a),(c)

 $\}^{2(a),(c)}$

2(a),(c)

Table 8-2 Modal analysis results for the single bay wet NPM model

ANSYS 3D MODEL	
X-Freq. (Hz)	X-Eff. Mass (lb·s ² /in)
11	
	11 ^{2(a),(c)}

ANSYS 3D MODEL	
Z-Freq. (Hz)	Z-Eff. Mass (lb·s ² /in)
11	
	11 ^{2(a),(c)}

ANSYS 3D MODEL	
Y-Freq. (Hz)	Y-Eff. Mass (lb·s ² /in)
11	
	11 ^{2(a),(c)}

$$\underbrace{\quad}_{\text{2(a),(c)}}$$

2(a),(c)

ANSYS 3D WET MODEL				
<u>Z-Freq.</u> <u>(Hz)</u>	<u>Z-Eff. Mass</u> <u>(lbf -s²/in)</u>	<u>Major NPM subcomponent</u> <u>responding at frequency</u>	<u>Type of mode</u>	<u>Modal mass to</u> <u>total mass (%)</u>
<u>}}</u>				

}}2(a),(c)

11

11^{2(a),(c)}

Figure 8-1 Wet model, 1st Significant Horizontal Mode in E-W Direction, 11

11^{2(a),(c)}

11

2(a),(c)

Figure 8-2 Wet model, 1st Significant Horizontal Mode in N-S Direction, 11

2(a),(c)

Location ID	Description	X (in)	Y (in)	Z (in)	Figure
31	{{			$\}}^{2(a),(c)}$	Figure A-7
32	{{			$\}}^{2(a),(c)}$	Figure A-2
33	{{			$\}}^{2(a),(c)}$	Figure A-2
34	{{			$\}}^{2(a),(c)}$	Figure A-2
35	{{			$\}}^{2(a),(c)}$	Figure A-2
36	{{			$\}}^{2(a),(c)}$	Figure A-2
37	{{			$\}}^{2(a),(c)}$	Figure A-2
38	{{			$\}}^{2(a),(c)}$	Figure A-1
39	{{			$\}}^{2(a),(c)}$	Figure A-1
40	{{			$\}}^{2(a),(c)}$	Figure A-1
41	{{			$\}}^{2(a),(c)}$	Figure A-1

8.4.2.2 Displacements, Accelerations, Rotations, and Relative Displacements

For each of the node points listed in Table 8-3, the displacements, accelerations, and rotations are extracted from the model at every time step for each direction in the global coordinate system.

8.4.2.3 Forces and Moments at Component Interfaces

The 3D NPM model is a “global” model that is used to compute internal load distributions, reaction forces and accelerations that are used to define the seismic loading applied for refined stress analysis of individual components using “local” finite-element analysis or classical methods. For this purpose, the resultant internal force and moment acting upon cross sections across shell structures or upon interfaces are evaluated as follows:

1. Nodal forces and moments associated with elements adjacent to a cross section or interface are calculated for all time points.
2. Elements on one side of the cross section-cut or interface are selected for following steps. The associated nodes on the cross section or interface are selected.
3. Forces and moments acting on the selected set of nodes from the selected elements are summed about a point at the centerline of the cross section or interface. Only forces and moments acting on the selected nodes and elements contribute to the resultant. For each time point, the resultant three force components and three moment components are stored.
4. For each force and moment component direction, the maximum absolute value is determined. Maximum forces and moments are summarized in a seismic loading specification for use in subsequent analysis. Note that the maximum values may occur at different times and from different NPM seismic analysis runs. Time histories of each force and moment component may be used for detailed analysis when necessary.

Forces and moments were generated for 77 interfaces between NPM components. Seven representative component interface locations are listed in Table 8-4. For the RPV upper support interfaces, remote points at the interface representing the center of the bolts are used instead of the center coordinates of the bolt hole surfaces.

Table 8-4 List of representative component interfaces for force and moment generation

Component Interface ID	Name	X (in)	Y (in)	Z (in)	Coordinate System	Figure
1	{{				$\}}^{2(a),(c)}$	Figure A-5
4	{{				$\}}^{2(a),(c)}$	Figure A-6
5	{{				$\}}^{2(a),(c)}$	Figure A-6
6	{{				$\}}^{2(a),(c)}$	Figure A-6
7	{{				$\}}^{2(a),(c)}$	Figure A-6
18	{{				$\}}^{2(a),(c)}$	Figure A-7
19	{{				$\}}^{2(a),(c)}$	Figure A-7

8.4.2.4 Forces and Moments within Component Sections

Forces and moments were generated for 22 internal sections of NPM components, including various elevations of the RPV, CNV, and RVI. Resultant forces and moments acting on internal components are evaluated as described in Section 8.4.2.3. The Section locations are listed in Table 8-5. Appendix A provides figures identifying the representative locations.

Table 8-5 List of component sections for force and moment generation

Component Section ID	Name	Elevation, Y (in)	Figure
1	{{	$\}}^{2(a),(c)}$	Figure A-5
2	{{	$\}}^{2(a),(c)}$	Figure A-5
3	{{	$\}}^{2(a),(c)}$	Figure A-5
4	{{	$\}}^{2(a),(c)}$	Figure A-5
5	{{	$\}}^{2(a),(c)}$	Figure A-5
6	{{	$\}}^{2(a),(c)}$	Figure A-5
7	{{	$\}}^{2(a),(c)}$	Figure A-5
8	{{	$\}}^{2(a),(c)}$	Figure A-5
9	{{	$\}}^{2(a),(c)}$	Figure A-6
10	{{	$\}}^{2(a),(c)}$	Figure A-6
11	{{	$\}}^{2(a),(c)}$	Figure A-6
12	{{	$\}}^{2(a),(c)}$	Figure A-6
13	{{	$\}}^{2(a),(c)}$	Figure A-6
14	{{	$\}}^{2(a),(c)}$	Figure A-6
15	{{	$\}}^{2(a),(c)}$	Figure A-6
16	{{	$\}}^{2(a),(c)}$	Figure A-6

Figure 8-9 ~~Example for in-structure response spectra broadening~~ Design ISRS, CNV Top Head, Z-Direction (North-South), 4% Damping

8.4.2.6 Maximum Uplift Displacements

The module vertical displacement was recorded at the CNV skirt (Table 8-3 Location 1) relative to the pool floor from the seismic model result files. The relative vertical displacement is the uplift of the module during seismic events. The maximum uplift displacement and occurring time were recorded for each case.

8.4.3 NuScale Power Module Seismic Analysis Results

Displacement and acceleration time-histories, maximum relative displacements, and broadened response spectra were generated for 33 locations in the NPM model. An additional set of relative displacements were generated for 8 more nodes on the RPV upper support to determine how much radial sliding occurs between each support

10.0 References

10.1 Referenced Documents

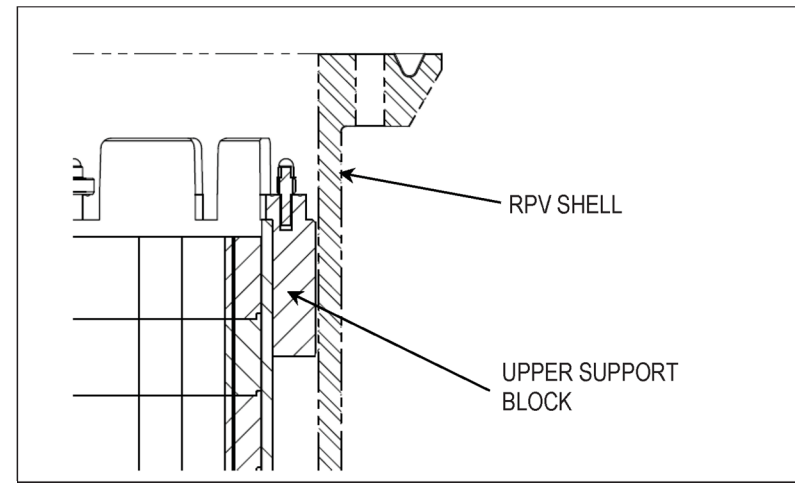
- 10.1.1 ASCE 4-13, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," Working Group on Revision of ASCE Standard 4, July 2013.
- 10.1.2 US NRC, "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications," DC/COL-ISG-001.
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- 10.1.11 Matthew D. Snyder, "Method for Hydrodynamic Coupling of Concentric Cylindrical Shells and Beams," 2004 International ANSYS Conference, Pittsburgh, PA, May 24-26, 2004.

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Figure C-7 CRDM support frame attached to the Containment vessel head (top view)

{{

Figure C-8 CRDM support frame attached to the RPV



}}^{2(a),(c)}

Figure C-9 Core support block with the lower core plate

Figure C-10 Core support upper support blocks (section view)

{{

}}2(a),(c)

Figure C-15 Upper riser hanger structural fastener (section view)

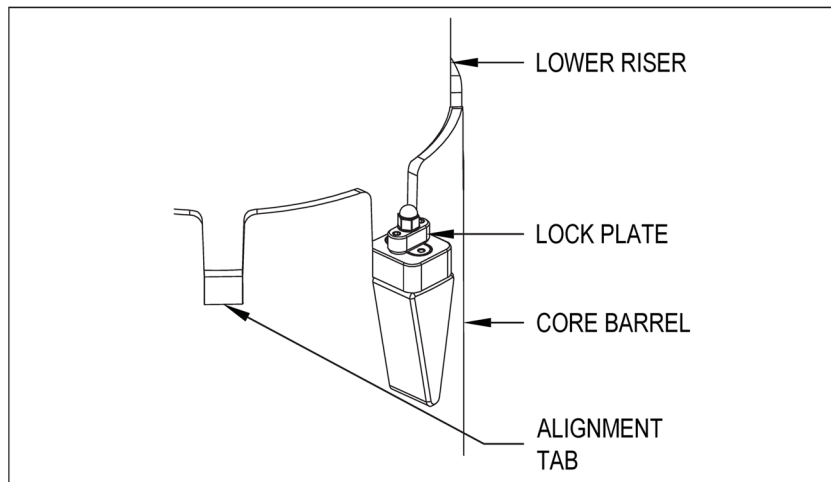


Figure C-17 Interface between lower riser and core support assembly

Figure C-16 Stacked reflector blocks above core plate (section view)

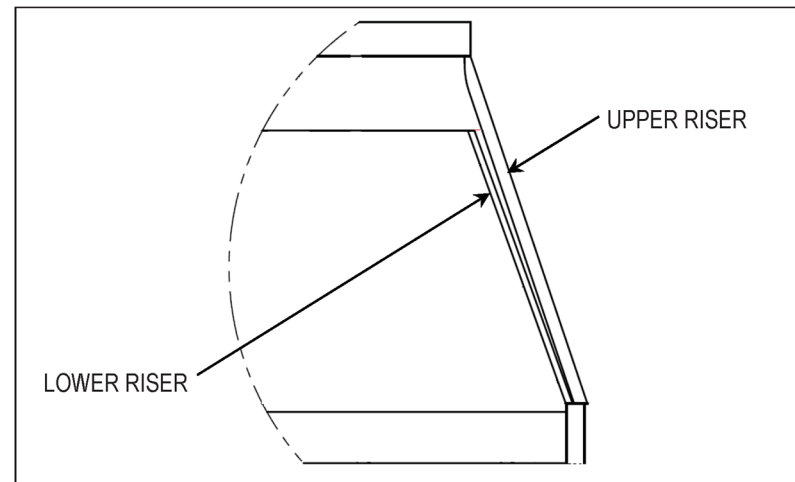


Figure C-18 Interface between upper and lower riser

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-47

10 CFR 50, Appendix A, GDC 4 requires structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The Standard Review Plan (SRP) establishes criteria that the NRC considers acceptable to use in implementing the agency's regulations. SRP 3.9.2, Rev. 3 states that initial piping startup testing should be performed in accordance with ASME OM-S/G-1990, "Standards and Guides For Operation of Nuclear Power Plants," Part 3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems."

DCD Tier 2, Section 3.9.2.1.1.1 states that ASME Code Class 1, 2, and 3 piping systems that are part of the reactor module are included within the scope of the NuScale Comprehensive Vibration Assessment Program (CVAP). This section also states that comprehensive vibration testing of all ASME Class 1, 2, and 3 piping is not required.

DCD Tier 2, Section 3.9.2.1.1.2 states that for ASME Code Class 3 piping that is not part of the reactor module (there is no Code Class 1 or 2 piping which is not part of the reactor module) and other ASME B31.1 piping outside of containment which requires vibration testing, vibration test specifications are developed in accordance with ASME OM-S/G, Division 2 (OM Standards), Part 3.

The CVAP piping system vibration measurement as discussed in RG 1.20, Rev. 3 is related to monitoring piping systems to ensure that vibration of the piping systems will not cause adverse effects for the reactor internals components. RG 1.20 does not provide acceptance criteria for piping. The piping vibration testing and the acceptance criteria should follow ASME OM S/G Part 3 provisions. The NRC staff requests that the applicant revise the application to perform the vibration testing in accordance with ASME OM S/G Part 3 for all piping systems within the scope of Section 3.9.2.

NuScale Response:

For traditional light water reactors, the scope of the comprehensive vibration assessment program (CVAP) is limited to reactor vessel internals only. NuScale took an alternate approach in defining both the scope of components and flow induced vibration mechanisms assessed in the NuScale CVAP. This is due in part to the integral nature of the NuScale reactor module, but more importantly is based on the desire to perform a complete vibration analysis, measurement, and testing program to design-out the risk of flow induced vibration (FIV) for the NuScale Power Module (NPM) components that are exposed to flow. NuScale believes that this approach provides a high degree of safety for components analyzed in the CVAP. Piping sections within the NPM considered susceptible to FIV are also included in the NuScale CVAP, as these piping segments constitute safety-related primary and secondary coolant pressure boundaries. There are industry examples of piping geometries that have experienced FIV; therefore, NuScale considered it prudent to include these piping regions in the CVAP, in accordance with Section 2 of Regulatory Guide (RG) 1.20, Rev. 3. Considering the integral nature of the NPM design, it is not desirable to wait until ASME OM-2012 testing to confirm that there are no “adverse” conditions in the NPM piping design.

Further, RG 1.20 provides guidance on the programmatic framework for performing vibration analysis and validation of the vibration analysis via the measurement and inspection programs. It can, therefore, be applied to any NuScale Power Module component (including piping) deemed susceptible to FIV.

Although the NPM piping is being analyzed, measured and inspected under the CVAP, NuScale also agrees that the requirements in Part 3 of ASME OM-2012, Division 2 (OM Standards) can be followed for piping that requires testing per the NuScale CVAP. FSAR Section 3.9.2.1.1.1 and CVAP TR-0716-50439 have been revised to note that the requirements in Part 3 of ASME OM-2012, Division 2 (OM Standards) are followed for the NPM piping that is required to be measured per the results of the CVAP analysis program.

NuScale additionally notes that the statement, *“The CVAP piping system vibration measurement as discussed in RG 1.20, Rev. 3 is related to monitoring piping systems to ensure that vibration of the piping systems will not cause adverse effects for the reactor internals components”* requires clarification based on the NuScale CVAP. The limiting safety margin for piping assessed in the CVAP was determined to be the decay heat removal steam piping. This piping is part of the secondary coolant pressure boundary and is located outside of the containment vessel. If vibration were experienced in this piping, it has no effect on the reactor vessel internals. Consequently, the NuScale CVAP does not base the acceptability of a flow induced vibration on whether it affects a reactor vessel internal. Additionally, while the statement, *“RG 1.20 does not provide acceptance criteria for piping”* is true, it should be clarified that RG 1.20 also does not provide acceptance criteria for reactor vessel internals, steam dryers, or other components specifically mentioned in RG 1.20. NuScale considered it appropriate to include the



NPM piping in the scope of the CVAP because RG 1.20 provides programmatic guidance to ensure that susceptible components in a prototype design do not experience unacceptable flow induced vibrations.

Impact on DCA:

FSAR Section 3.9.2.1 and the CVAP TR-0716-50439 Section 4.3 have been revised as described in the response above and as shown in the markup provided with this response.

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, then corrective actions (i.e. reanalyzing with as-built values) are implemented and the systems are retested.

Phase II - Initial Startup Testing

Initial startup testing is performed after the reactor core is loaded into a reactor module. These Phase II tests establish that the vibration level and piping reactions to transient conditions are acceptable and bounded by the analyses. If the vibration levels are not bounded, the analyses use the vibration level from the testing as input and verify that the design is acceptable.

3.9.2.1.1 Piping Vibration Details

3.9.2.1.1.1 Piping Included in Comprehensive Vibration Assessment Program

RAI 03.09.02-47

ASME Code Class 1, 2, and 3 piping systems that are part of the reactor module are included within the scope of the NuScale Comprehensive Vibration Assessment Program (CVAP) (Reference 3.9-5) Piping systems that meet the screening criteria for applicable flow induced vibration mechanisms are evaluated in the analysis program. If a large margin of safety is not demonstrated, prototype testing is performed in accordance with the CVAP measurement program [and the requirements of Part 3 of ASME OM-2012, Division 2 \(OM Standards\)](#).

RAI 03.09.02-47

Reactor module components, piping, and supports with a high degree of safety margin are excluded from testing in the prototype measurement program, consistent with the overall measurement program objectives of validating relevant analytical inputs, results, and margins of safety. ~~Therefore, comprehensive vibration testing of all ASME Code Class 1, 2, and 3 piping is not performed.~~

3.9.2.1.1.2 Piping Not Included in Comprehensive Vibration Assessment Program

For ASME Code Class 3 piping that is not part of the reactor module (there is no Code Class 1 or 2 piping which is not part of the reactor module) and other ASME B31.1 piping outside of containment which requires vibration testing, vibration test specifications are developed in accordance with ASME OM-S/G, Division 2 (OM Standards), Part 3 (Reference 3.9-3). SRP 3.9.2 recommends using this part of the ASME OM Code for developing preoperational vibration test specifications. Piping vibration testing and assessment are performed in accordance with ASME OM-2012, Division 2 (OM Standards), Part 3 (Reference 3.9-3).

~~is performed to provide additional assurance that sufficiently bounding inputs have been used in the analysis.~~

~~The scope of the separate effects testing is to measure vibration amplitudes of the CRA fingers and rodlets as a function of inlet flow velocity, which provides data for validating the response of these structures to TB and VS. This test is being performed because it is recognized that the CRA GT represents a unique region of the RVI that may not be best characterized using existing literature approaches. Vibration measurements are taken at five flow rates up to 150 percent of the full-power primary-coolant flow rate.~~

4.2 Lead Unit Factory Testing

During the factory testing phase, testing is performed to verify component natural frequencies. Due to the natural circulation design of the NPM, it is not possible to perform flow testing without using temporary systems to provide the required primary and secondary-flow conditions necessary to validate portions of the analysis program. Design and installation of temporary systems to achieve full-power flow conditions prior to initial startup testing is impractical. Because the NPM components are subjected to low velocities characteristic of natural circulation and there are large factors of safety for susceptible components, flow testing of NPM components is not performed prior to the initial startup test phase.

4.2.1 In-Air Component Frequency Testing

In-air frequency tests are performed on the prototype NPM to determine the natural frequencies of the CRDS and ICIGT. The fundamental frequency for these components requires verification in order to justify the margin obtained in the VS analysis. Because damping is not used in VS analysis of these components and hydrodynamic mass can be approximated analytically using accepted empirical correlations, it is acceptable to validate the fundamental frequencies of the prototype using in-air testing.

4.3 Lead Unit Initial Startup Testing

Initial startup testing is performed on the first NPM after the first fuel load. Due to the natural circulation design of the NPM, it is not possible to obtain the limiting TH conditions that are necessary to verify the FIV inputs and results until the NPM is operating near full-power conditions. Initial startup testing will be performed for a sufficient duration to ensure one million vibration cycles for the component with the lowest structural natural frequency. It is expected to take less than two days to obtain one million cycles of vibration. This is a conservative estimate because the lowest natural frequency of any component evaluated in the CVAP is approximately $\{ \{ \}^{2(a),(c),ECI}$.

The initial startup test will be performed with online vibration monitoring of the DHRS steam piping. Testing of this piping section is performed in accordance with the requirements of Part 3 of ASME OM-2012, Division 2 (OM Standards). In the event that an unacceptable vibration response develops any time during initial startup testing, the test conditions will be adjusted to stop the vibration and the reason for the vibration

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-48

10 CFR 52.47 requires design certification applicants to demonstrate how operating experience insights have been incorporated into the plant design. Operating experience indicates that acoustic resonance at closed branching piping can cause acoustic resonance which leads to excessive vibration that can damage piping system components. Minor changes in piping design such as the branch transition radius can cause adverse vibration in the piping system. The NRC staff requests that the applicant provide a discussion in DCD Tier 2, Section 3.9.2.1 the process that is used to design all ASME Code Class 1, 2, and 3 piping systems to avoid vibration phenomena such as acoustic resonance at pipe branches. Without the requested information, the staff cannot reach a safety finding.

NuScale Response:

The NuScale Comprehensive Vibration Assessment Program (CVAP) addresses flow induced vibration (FIV) phenomena, including acoustic resonance, for the NPM. The CVAP includes predictive analysis for FIV mechanisms, as well as testing and inspections for validation of the analytical methods and results. The CVAP is described in detail in “NuScale Comprehensive Vibration Assessment Program Technical Report,” TR-0716-50439. The scope of the CVAP includes reactor vessel internals and structures, steam generator components, and the NPM piping. As discussed in FSAR Section 3.9.2.1.1.1, all ASME Code Class 1 and 2 piping, and part of the Class 3 piping, is within the NPM, and therefore, is within the scope of the CVAP.

As a result of the predictive analysis performed for the CVAP, design features of the NPM are included in order to preclude vibration due to acoustic resonance at closed pipe branches. The NuScale design includes:

- widening of the inlets of the decay heat removal system (DHRS) branch lines off the feedwater (FW) lines,
 - and offsetting the DHRS lines on the MS lines to avoid a coaxial configuration.
-



As discussed in FSAR Section 3.9.2.1.1.2, part of the ASME Code Class 3 piping is located beyond the NPM, and therefore, is not within the scope of the CVAP. This piping is nonsafety-related and the closed branches on the piping includes vent and drain lines and instrument tubing only. Compliance with ASME OM-2012, Division 2, Part 3, addresses vibration concerns in this piping and all piping beyond the NPM. Per ASME OM-2012, this ASME Code Class 3 piping requires steady state and transient vibration testing. Part 3 recognizes the phenomena of acoustic resonance due to vortex shedding at branch lines, and provides precautions that branch piping should be given attention. If adverse piping vibration is discovered during this testing, corrective measures are implemented. Also, there is no operating experience that suggests acoustic resonance of small branches (e.g., \leq NPS 1 vents, drains, and instrument tubes), such as those on the ASME Code Class 3 piping beyond the NPM, could cause adverse acoustic loading on safety-related components that are remote from the branch itself (e.g., the reactor vessel or reactor vessel internals). No additional information is required in the FSAR to address acoustic resonance of ASME Code Class 1, 2, and 3 piping systems.

There is operating experience that acoustic resonance of closed branches in large steam piping remote from the reactor vessel could cause adverse acoustic loading on the reactor vessel or reactor vessel internals. NRC Information Notice IN-2002-26, including Supplements 1 & 2, describes fatigue failures of the steam dryers in Quad Cities Unit 1 and 2 BWRs. Later evaluations determined that the failures were caused by acoustic resonance in the main steam line relief valve standpipes. The NPS 12 MS lines in the NuScale design include NPS 4 bypass lines around the secondary main steam isolation valves (MSIVs). During normal operation, the bypass valves are closed, and the bypass lines are closed branches off of the MS lines. This configuration could experience acoustic resonance, similar to that of the Quad Cities events, unless the branch lines are designed to preclude the phenomenon.

Section 3.9.2.1.1.3 has been added to the FSAR to address the concern of MS line branch piping acoustic resonance. However, as mentioned in the response to RAI 8942 Question No. 03.06.02-15, NuScale has used a 'graded level of detail' approach for piping design. Therefore, the routing of the remainder of the piping beyond the Reactor Pool Wall, including the design of the bypass lines around the secondary MSIVs, is to be completed by the COL applicant as described in COL Item 3.6-1. A new COL Item 3.9-10 has been added to ensure that the phenomenon of acoustic resonance is considered in the detailed design of the MS line beyond the Reactor Bay Wall.

Impact on DCA:

FSAR Section 3.9.2.1 and Table 1.8-2 have been revised as described in the response above and as shown in the markup provided with this response.

RAI 02.04.13-1, RAI 03.04.02-1, RAI 03.04.02-2, RAI 03.04.02-3, RAI 03.05.01.04-1, RAI 03.05.02-2, RAI-03.06.02-15, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.09.02-15, RAI 03.09.02-48, RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-6, RAI 03.09.06-16, RAI 03.09.06-27, RAI 03.11-8, RAI 03.11-14, RAI 06.04-1, RAI 09.01.05-3, RAI 09.01.05-6, RAI 09.03.02-3, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8, RAI 10.02-1, RAI 10.02-2, RAI 10.04.10-2, RAI 13.01.01-1, RAI 13.01.01-1S1, RAI 13.02.02-1, RAI 13.03-4, RAI 13.05.02.01-2, RAI 13.05.02.01-2S1, RAI 13.05.02.01-3, RAI 13.05.02.01-3S1, RAI 13.05.02.01-4, RAI 13.05.02.01-4S1

Table 1.8-2: Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.	1.1
COL Item 1.1-2:	A COL Applicant applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.	1.7
COL Item 1.7-2:	A COL Applicant applicant that references the NuScale Power Plant design certification will list additional site-specific P&IDs and legends as applicable.	1.7
COL Item 1.8-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.	1.8
COL Item 1.9-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of any management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.	1.10
COL Item 2.0-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will demonstrate that site-specific characteristics are bounded by the design parameters specified in Table 2.0-1. If site-specific values are not bounded by the values in Table 2.0-1, the COL applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its combined license application.	2.0
COL Item 2.1-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each potential accident, or provide site-specific design alternatives.	2.2
COL Item 2.3-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will describe the site-specific meteorological characteristics for Section 2.3.1 through Section 2.3.5, as applicable.	2.3
COL Item 2.4-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will investigate and describe the site-specific hydrologic characteristics for Section 2.4.1 through Section 2.4.14, as applicable.	2.4
COL Item 2.5-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will describe the site-specific geology, seismology, and geotechnical characteristics for Section 2.5.1 through Section 2.5.5, below.	2.5

Table 1.8-2: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.9-10:	<u>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.</u>	<u>3.9</u>
COL Item 3.10-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will develop and maintain a site-specific seismic and dynamic qualification program.	3.10
COL Item 3.10-2:	A COL Applicant applicant that references the NuScale Power Plant design certification will develop the equipment qualification database and ensure equipment qualification record files are created for the structures, systems, and components that require seismic qualification.	3.10
COL Item 3.10-3:	A COL Applicant applicant that references the NuScale Power Plant design certification will submit an implementation program for Nuclear Regulatory Commission approval prior to the installation of the equipment that requires seismic qualification.	3.10
COL Item 3.11-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will submit a full description of the Environmental Qualification Program and milestones and completion dates for program implementation.	3.11
COL Item 3.11-2:	A COL Applicant applicant that references the NuScale Power Plant design certification will develop the equipment qualification database and ensure equipment qualification record files are created for the structures, systems, and components that require environmental qualification.	3.11
COL Item 3.11-3:	<u>A COL applicant that references the NuScale Power Plant design certification will implement an EQ operational program that incorporates the above aspects specific to the EQ of mechanical and electrical equipment.</u>	<u>3.11</u>
COL Item 3.11-4:	<u>A COL applicant that references the NuScale Power Plant design certification will ensure the Environmental Qualification Program cited in COL Item 3.11-1 includes a description of how equipment located in harsh conditions will be monitored and managed throughout plant life. This description will include methodology to ensure equipment located in harsh environments will remain qualified if the measured dose is higher than the calculated dose.</u>	<u>3.11</u>
COL Item 3.12-1:	A COL Applicant applicant that references the NuScale Power Plant design certification may use a piping analysis program other than the programs listed in Section 3.12.4.1; however, the applicant will implement a benchmark program using the models for NuScale Power Plant standard design.	3.12
COL Item 3.12-2:	A COL Applicant applicant that references the NuScale Power Plant design certification will confirm that the site-specific seismic response is within the parameters specified in Section 3.7. A COL applicant may perform a site-specific piping stress analysis in accordance with the methodologies described in this section, as appropriate.	3.12
COL Item 3.13-1:	A COL Applicant applicant that references the NuScale Power Plant design certification will provide an inservice inspection program for ASME Class 1, 2 and 3 threaded fasteners or describe the implementation program, including milestones, completion dates and expected conclusions. The program will identify the applicable edition and addenda of ASME BPVC, Section XI and ensure compliance with 10 CFR 50.55a.	3.13
COL Item 5.2-1:	A COL Applicant applicant that references the NuScale Power Plant design certification and uses a later Code edition or addenda other than American Society of Mechanical Engineers Boiler and Pressure Vessel Code 2013 will perform and document with a code reconciliation an American Society of Mechanical Engineers Design Report as required by American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Paragraph NCA-3554, "Modification of Documents and Reconciliation With Design Report."	5.2

observed which is outside the bounds of the analyses, a re-analysis to determine the cause and to identify the corrective action is performed.

Service Level B loads are infrequent loads with a high probability of occurrence but which cause no damage or reduction in component function. The vibrations are the result of valve operation, pumps, and other loads from transients. If excessive vibration is observed which is outside the bounds of the analyses, a re-analysis to determine the cause and to identify the corrective action is performed.

The Phase I and Phase II tests do not address vibrations resulting from Service Level C or Service Level D loads.

RAI 03.09.02-48

3.9.2.1.1.3

Main Steam Line Branch Piping Acoustic Resonance

NRC Information Notice IN-2002-26, including Supplements 1 & 2, describes fatigue failures of steam dryers in BWRs, which occurred at Quad Cities Units 1 and 2. Later evaluations determined that the failures were caused by acoustic resonance in the main steam line relief valve standpipes. The NuScale design MS lines (NPS 12) include bypass lines (NPS 4) around the secondary main steam isolation valves. During normal operation, the bypass valves are closed, and the bypass lines are closed branches off of the MS lines. This configuration is similar to that of the Quad Cities events. Therefore, 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.

RAI 03.09.02-48

COL Item 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.

3.9.2.1.2

Piping Thermal Expansion Details

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. The tests also provide verification that the component supports can accommodate the expansion of the piping for the service levels for these modes of operation. Section 14.2 provides descriptions of selected planned piping thermal expansion measurement tests. Test specifications for thermal expansion testing of piping systems during preoperational and start-up testing will be in accordance with ASME OM Standard (Reference 3.9-3), Part 7.

Response to Request for Additional Information Docket No. 52-048

eRAI No.: 8911

Date of RAI Issue: 08/25/2017

NRC Question No.: 03.09.02-49

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. DCD Tier 2, Rev. 0, Section 3.7.1, Table 3.7.1-6 contains SSE damping values for NPM dynamic analysis. Damping values for structural materials, piping systems, electrical distribution systems, and HVAC Duct Systems are presented. The damping values of these systems are consistent with the damping values specified in RG 1.61, Revision 1, except the damping value of the sloshing mode of metal atmospheric storage tanks. Table 3.7.1-6 specifies SSE damping value of 2% for the sloshing mode of metal atmospheric storage tank which is not consistent with the damping value of 0.5% for sloshing mode of metal atmospheric storage tank specified in RG. 1.61, Revision 1. The NRC staff requests that the applicant provide the justification for using a lower damping value, and update the DCD to include the justification. Without the requested information, the staff cannot reach a safety finding.

NuScale Response:

FSAR Section 3.7.1, Table 3.7.1-6 has been updated to correct the SSE and OBE damping values for the Metal Atmospheric Storage Tanks to be consistent with RG 1.61 Revision 1, Table 6. The NRC was notified of this FSAR revision by NuScale letter to U.S. Nuclear Regulatory Commission, "NuScale Power, LLC (NuScale) Submittal of Changes to Final Safety Analysis Report, Section 3.7, Seismic Design," dated July 14, 2017 (page 5 of 37).

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Enclosure 3:

Affidavit of Zackary W. Rad, AF-1017-56807

NuScale Power, LLC
AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.
2. I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:
 - a. The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.
 - b. The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.
 - c. Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.
 - d. The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.
 - e. The information requested to be withheld consists of patentable ideas.
3. Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the method and analyses by which NuScale develops its power module seismic analysis.

NuScale has performed significant research and evaluation to develop a basis for this method and analyses and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

4. The information sought to be withheld is in the enclosed response to NRC Request for Additional Information No. 202, eRAI No. 8911. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.
5. The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).
6. Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:
 - a. The information sought to be withheld is owned and has been held in confidence by NuScale.
 - b. The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.
 - c. The information is being transmitted to and received by the NRC in confidence.
 - d. No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.
 - e. Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 10/24/2017.

A handwritten signature in black ink, appearing to read 'Zackary W. Rad', is written over a horizontal line.

Zackary W. Rad