June 08, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

**SUBJECT:** NuScale Power, LLC Response to NRC Request for Additional Information No. 410 (eRAI No. 9310) on the NuScale Design Certification Application

**REFERENCE:** U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 410 (eRAI No. 9310)," dated April 09, 2018

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. 9310:

- 03.09.02-59
- 03.09.02-60
- 03.09.02-61
- 03.09.02-63
- 03.09.02-66
- 03.09.02-67
- 03.09.02-68
- 03.09.02-72

The schedule for questions 03.09.02-62, 03.09.02-64, 03.09.02-69, 03.09.02-70 and 03.09.02-71 was provided in emails to NRC (Greg Cranston) dated May 09, 2018 and June 1, 2018. The response to question 03.09.02-65 will be provided by June 22, 2018.

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 410 (eRAI No. 9310). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response. 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.

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
Marieliz Vera, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9310, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9310, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0618-60343

**Enclosure 1:**

NuScale Response to NRC Request for Additional Information eRAI No. 9310, proprietary

**Enclosure 2:**

NuScale Response to NRC Request for Additional Information eRAI No. 9310, nonproprietary

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

**eRAI No.:** 9310
**Date of RAI Issue:** 04/09/2018

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**NRC Question No.:** 03.09.02-59

In the response to RAI 8911, Question 03.09.02-19, the applicant stated that the results of the NuScale power module (NPM) seismic analysis bound the results of other NPM locations including the NPM placed in the containment flange tool (CFT) and reactor flange tool (RFT), as well as suspended by reactor building crane. The NPM seismic analysis in dry dock was not performed because the NPM is classified as Seismic Category III structure while in dry dock. For the NPM in the transition mode, the applicant stated that during NPM transport, the NPM is isolated from the transmission of horizontal seismic loads because the module lifting adapter does not provide lateral restrain when suspended by the building crane. The staff finds that the RAI response is incomplete. The applicant is requested to address the NPM seismic response under vertical seismic loading when suspended by the building crane.

Include the requested information in the NPM Seismic Report TR-0916-51502.

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**NuScale Response:**

Seismic response of the NuScale Power Module (NPM) components in refueling transition modes (i.e., when suspended by the building crane) is not evaluated. As previously cited, the NPM in transition is isolated from the transmission of horizontal seismic loads because the module lifting adapter does not provide lateral restraint when suspended by the building crane. Evaluation of vertical seismic loads is not performed because the NPM is suspended by the building crane only for short durations of time.

The movements of the NPM and associated durations taken from current operations procedures are listed in the table below.

| Crane Movement | Duration hr |
|---|---|
| **Operating bay to CNV flange tool** | |
| Lift and move module to the CNV flange tool | 3 |
| Align the module to the CNV flange tool | 1 |
| Visually inspect containment externals | 1 |
| Lower the module into the CNV flange tool | 1 |
| **CNV flange tool to RPV flange tool** | |
| Visually inspect upper and lower containment flanges | 1 |
| Visually inspect lower RPV externals | 1 |
| Move module to RPV flange tool | 1 |
| Align the module to the RPV flange tool | 1 |
| Lower the module into the reactor flange tool | 1 |
| **RPV flange tool to CNV flange tool** | |
| Move upper module to the CNV flange tool | 1 |
| Align the module to the CNV flange tool guides | 1 |
| Lower module to establish a 4” flange gap | 1 |
| Using containment tool camera, inspect flanges | 1 |
| Align upper containment flange to lower -then set | 1 |
| **CNV flange tool to operating bay** | |
| Move the assembled module to the operating bay | 2 |
| Secure the module and release the reactor building crane | 1 |
| Total: | 19 hours |

The refueling cycle for each NPM is two years. For a twelve module plant, refueling of six modules per year is anticipated. Therefore, this refueling transition configuration exists for 4.75 days per year.

This time in transition is much less than the refueling outage durations for the U.S. nuclear operating fleet. Operating experience outage details (as tabulated for the U.S. nuclear operating fleet for calendar years 2000 through 2017) document average refueling durations (excluding extended outages over 125 days) of 40.2 days for all Pressurized Water Reactors, 34.6 days for all Boiling Water Reactors, and 38.5 days for all plants. The operating nuclear fleet does not perform seismic evaluation of the reactor components and nuclear fuel during refueling operations based on their short time durations in the refueling phases.

Control and governance of the Reactor Building Crane operation with heavy loads (including movement of the NPM) is addressed in the NuScale FSAR Section 9.1.5, ‘Overhead Heavy Load Handling Systems.’ Combined License (COL) Item 9.1-5 and COL Item 9.1-7 specify the following:

COL Item 9.1-5: The COL applicant that references the NuScale Power Plant design certification will describe the process for handling and receipt of critical loads including NuScale Power Modules.

COL Item 9.1-7: The COL applicant that references the NuScale Power Plant design certification will provide a description of the program governing heavy loads handling. The program should address

• operating and maintenance procedures

• inspection and test plans

• personnel qualification and operator training

COL applicant processes for heavy load handling control the lift durations and preclude the ‘parking’ of apparatus in critical load configurations.

**Impact on DCA:**

TR-0916-51502 Section 1.2 has been revised as described in the response above and as shown in the markup provided with 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.

The NPM seismic analysis applies to the NPM while situated in the operating bays and while secured in the refueling tool. Refueling transitions, such as the NPM suspended by the building crane, are not seismically evaluated, based on the short time durations in these configurations. For a twelve module NuScale power plant, the time in refueling transition configurations is less than five days per year (assuming a two year fuel cycle per NPM, the refueling of six modules per year, and an estimated 20 hours of crane movement per NPM refueling). This 'time in refueling transition' is less than the refueling outage durations for the U.S. nuclear operating fleet. The operating fleet does not perform seismic evaluation of the reactor components and nuclear fuel during refueling operations based on their short duration of time in the refueling phases.

Operating experience outage details as tabulated for the U.S. nuclear operating fleet (calendar year 2000 through 2017), document average refueling durations (excluding extended outages) of 40.2 days for all Pressurized Water Reactors, 34.6 days for all Boiling Water Reactors, and 38.5 days for all nuclear plants.

COL Items 9.1-5 and 9.1-7 governing the handling of heavy loads ensure the durations are controlled in these refueling configurations.

## 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 |

# Response to Request for Additional Information
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**eRAI No.:** 9310
**Date of RAI Issue:** 04/09/2018

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**NRC Question No.:** 03.09.02-60

In the response to RAI 8911, Question 03.09.02-25, the applicant stated that the horizontal fluid masses in question are listed as “Core + reflector channels,” “Lower riser,” and “Riser transition” in Table 4-13 of NPM Seismic Report TR-0916-51502 and 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 of TR-0916-51502. Half of this mass is applied using vertical point masses to the inside surface of the lower reactor pressure vessel (RPV) head, and half is applied to the bottom of the pressurizer baffle plate. The staff requests to add this information in technical report TR-0916-51502.

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**NuScale Response:**

The information provided in the response to RAI 8911 Question 03.09.02-25 related to the application of horizontal and vertical fluid masses in the seismic evaluation of the Reactor Pressure Vessel has been added to Section 4.1.8.1 of TR-0916-51502 as requested.

**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.

#### 4.1.8.1 Reactor Coolant System Fluid Mass

The RCS volumes are summarized in Table 4-14. The reference volumes already account for space displaced by SG tubes and fuel assemblies, and therefore are not adjusted further. The annular volumes are taken from the Fluid Coupling (Section 4.1.8.5) for consistency within this model. The total RCS volume in this model is {{ }}$^{2(a),(c)}$. The volume region locations of Table 4-14 are depicted in Figure 4-22.

The masses are calculated by multiplying the volume of each region by the water density at the approximate average RCS temperature of 550 degrees F and pressure of 1850 psia, except for the PZR volumes. The two PZR regions use the density of liquid water and steam at saturation (for 1850 psia and 625 degrees F). The RCS volumes and masses are summarized in Table 4-14.

The mass of the fluid within the lower RVI is accounted for as follows. The horizontal fluid masses are listed as “Core + reflector channels,” “Lower riser,” and “Riser transition” in Table 4-14 and 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 4-14. Half of this mass is applied to the inside surface of the lower reactor pressure vessel (RPV) head, and half is applied to the bottom of the pressurizer baffle plate.

Table 4-14 Reactor coolant system volumes and fluid masses

| Region | Volume (in$^3$) | Density (lbm/in$^3$) | Mass (lbm) |
|---|---|---|---|
| Lower plenum | {{ | | }}$^{2(a),(c)}$ |
| Core + reflector channels | {{ | | }}$^{2(a),(c)}$ |
| Lower riser | {{ | | }}$^{2(a),(c)}$ |
| Riser transition | {{ | | }}$^{2(a),(c)}$ |
| Upper riser | {{ | | }}$^{2(a),(c)}$ |
| Upper riser supports | {{ | | }}$^{2(a),(c)}$ |
| PZR liquid (65%) | {{ | | }}$^{2(a),(c)}$ |
| PZR vapor (35%) | {{ | | }}$^{2(a),(c)}$ |
| Annular volume SG | {{ | | }}$^{2(a),(c)}$ |
| Annular volume lower | {{ | | }}$^{2(a),(c)}$ |
| **TOTAL** | {{ | | }}$^{2(a),(c)}$ |
| Main RCS Total (no PZR) | {{ | | }}$^{2(a),(c)}$ |
| PZR Total | {{ | | }}$^{2(a),(c)}$ |

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**Date of RAI Issue:** 04/09/2018

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**NRC Question No.:** 03.09.02-61

In the response to Subquestion 1 of RAI 8911, Question 03.09.02-26, the applicant stated that the total fuel mass of the fuel assembly beam-spring model does not included the fluid mass and the fluid mass determined in Table 4-14 of TR-0916-51502 is added separately to the NPM full model. The staff noticed that Table 4-14 of TR-0916-51502 contains fluid mass of steam generator only. The applicant is requested to provide the correct table/reference of fuel assembly fluid mass that was added to the NPM full model.

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

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**NuScale Response:**

The observation of an incorrect table number as cited in the RAI question is correct. Table 4-13 of the NPM Seismic Analysis technical report TR-0916-51502 Revision 0 is titled, “Reactor Coolant System Volumes and Fluid Masses” and includes the fluid mass for the region of the fuel assemblies in the row “Core + reflector channels.”

The cause for this numbering discrepancy is explained as follows:

The response to RAI 8911 submitted by NuScale letter, "Response to NRC Request for Additional Information No. 202 (eRAI No. 8911) on the NuScale Design Certification Application," dated October 24, 2017, included numerous technical report text, figure, and table revisions associated with the many RAI 8911 questions. This resulted in the re-numbering of the TR table and figure numbers. Specifically, the RAI 8911 response to Question 03.09.02-22 added a new Table 4-4. This effectively caused the subsequent table numbers to change (by +1) and the TR Revision 0 Table 4-13 to become Table 4-14 in the later RAI 8911 Question 03.09.02-26 response (same letter).

This response clarifies that NPM Seismic Analysis technical report TR-0916-51502 Revision 0 includes fluid masses in the Table 4-13, 'Reactor Coolant System Volumes and Fluid Masses.' However, it is noted that in the upcoming scheduled Revision 1 of this technical report, incorporating markups associated with the issued responses to RAI 8911 and RAI 9310, this table number will be indexed appropriately.

**Impact on DCA:**

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

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**Date of RAI Issue:** 04/09/2018

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**NRC Question No.:** 03.09.02-63

In the response to Subquestion 1 of RAI 8911, Question 03.09.02-28, the applicant stated that "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) and no friction coefficient is assigned to the elements. The no-separation contact is modelled by ANSYS CONTA174 and TARGE170 element. The staff noticed that the input of CONTA174 allows assignment of friction coefficient through material property parameter, MU. Friction in the joint can affect component natural frequency. Provide justification that omission of friction coefficient (i. e., frictionless contact) in the two lower reactor internals boundary conditions mentioned in this RAI provides conservative results.

Include the requested information in the NPM Seismic Report.

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**NuScale Response:**

No separation contact surfaces are used in the two lower reactor internals boundary conditions as discussed herein.

For these contact surfaces, the effect of additional energy dissipation due to Coulomb friction is neglected. The primary load path for resisting external seismic loads is through the contact forces normal to the the interfaces. By setting the coefficient of friction to zero, Coulomb friction is not used to provide additional resistance to external seismic loads through a secondary load path which can relieve forces acting on the primary load path. The exclusion of frictional resistance results in a conservative assessment of the normal seismic forces acting on the interface.

The natural frequency of the system is a characteristic of the linear system. Frictional forces render the dynamic response nonlinear and act to perturb the dynamic response. When friction is included, the response is not characterized by modal analysis or changes in component natural frequency. The effect of friction in perturbing the dynamic response is addressed by the inclusion of uncertainty in the analysis of the NPM dynamic response for a range of stiffness values. The inclusion of uncertainty in the dynamic analysis encompasses numerous causes of uncertainty, including the effect of neglecting friction at these interfaces.

Internal boundary between upper support blocks and the lower RPV shell:

The upper support blocks to RPV shell no-separation contact surfaces are shown on Figure 4-11 of the TR-0916-51502. Neglecting friction, the contact forces act only in the radial direction. The components slide vertically and circumferentially relative to each other due to relative deflections of the lower RPV and the lower RVI shells.

Transfer of load across this interface is assumed to occur through radial forces alone. Therefore, the primary load path and resistance provided by this interface is in the radial direction. Additional resistance due to frictional circumferential forces is neglected. In the vertical direction the primary load path is through the core barrel. The secondary support path due to vertical friction at the core support blocks is neglected.

Large dissipative Coulomb friction forces are expected. Energy dissipation due to Coulomb friction forces is neglected by setting the coefficient of friction to zero.

Internal boundary between lower RVI and reflector:

The lower RVI to reflector no-separation contact surfaces are shown on Figure 4-12 of the TR-0916-51502. Neglecting friction, the contact forces act only in the radial direction, therefore, the primary load path provided by this interface is in the radial direction.

The reflector blocks and core barrel are physically separated by a water filled annular gap. The fluid filled gap is nominally 0.125 inch thick. Coulomb friction does not act on the interface between the reflector and core barrel unless the gap closes due to relative displacements. Viscous forces acting due to the fluid do not cause significant damping and are conservatively neglected.

For horizontal seismic loads, the primary load path for forces acting on the reflector is through alignment pins between each stacked block and between the lowest block and lower core plate. A secondary load path for horizontal loads is through the action of inertial and viscous forces resulting from fluid within the annular gap. Additional discussion and justification for representing the fluid gap using a linear no-separation condition is to be provided in the response to RAI 9310 Question 03.09.02-70.

For vertical loads, the primary load path is between stacked blocks and the lower core plate. Forces due to Coulomb friction occur only if deflection is sufficient to close the gap and do not provide a primary load path for resisting vertical seismic load.

**Impact on DCA:**

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

{{

}}2(a),(c)

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.

No separation contact surfaces are used in the two lower reactor internals boundary conditions as discussed herein.

For these contact surfaces, the effect of additional energy dissipation due to Coulomb friction is neglected. The primary load path for resisting external seismic loads is through the contact forces normal to the interfaces. By setting the coefficient of friction to zero, Coulomb friction is not used to provide additional resistance to external seismic loads through a secondary load path, which can relieve forces acting on the primary load path. The exclusion of frictional resistance results in a conservative assessment of the normal seismic forces acting on the interface.

The natural frequency of the system is a characteristic of the linear system. Frictional forces render the dynamic response nonlinear and act to perturb the dynamic response. When friction is included, the response is not characterized by modal analysis or changes in component natural frequency. The effect of friction in perturbing the dynamic response is addressed by the inclusion of uncertainty in the analysis of the NPM dynamic response for a range of stiffness values. The inclusion of uncertainty in the dynamic analysis encompasses numerous causes of uncertainty, including the effect of neglecting friction at these interfaces.

Internal boundary between upper support blocks and the lower RPV shell:

The upper support blocks to RPV shell no-separation contact surfaces are shown on Figure 4-11. Neglecting friction, the contact forces act only in the radial direction. The components slide vertically and circumferentially relative to each other due to relative deflections of the lower RPV and the lower RVI shells.

Transfer of load across this interface is assumed to occur through radial forces alone. Therefore, the primary load path and resistance provided by this interface is in the radial direction. Additional resistance due to frictional circumferential forces is neglected. In the vertical direction the primary load path is through the core barrel. The secondary support path due to vertical friction at the core support blocks is neglected.

Large dissipative Coulomb friction forces are expected. Energy dissipation due to Coulomb friction forces is neglected by setting the coefficient of friction to zero.

Internal boundary between lower RVI and reflector:

The lower RVI to reflector no-separation contact surfaces are shown on Figure 4-12. Neglecting friction, the contact forces act only in the radial direction, therefore, the primary load path provided by this interface is in the radial direction.

The reflector blocks and core barrel are physically separated by a water filled annular gap. The fluid filled gap is nominally 0.125 inch thick. Coulomb friction does not act on the interface between the reflector and core barrel unless the gap closes due to relative displacements. Viscous forces acting due to the fluid do not cause significant damping and are conservatively neglected.

For horizontal seismic loads, the primary load path for forces acting on the reflector is through alignment pins between each stacked block and between the lowest block and lower core plate. A secondary load path for horizontal loads is through the action of inertial and viscous forces resulting from fluid within the annular gap.

For vertical loads, the primary load path is between stacked blocks and the lower core plate. Forces due to Coulomb friction occur only if deflection is sufficient to close the gap and do not provide a primary load path for resisting vertical seismic load.

{{

}}2(a),(c)

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

# Response to Request for Additional Information
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**eRAI No.:** 9310
**Date of RAI Issue:** 04/09/2018

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**NRC Question No.:** 03.09.02-66

In the response to Subquestion 2 of RAI 8911, Question 03.09.02-38, the applicant stated that the in-structure floor response spectra provided in the technical report TR-0916-51502 were generated using the guidance provided in RG 1.122, Rev. 1 and reduction of narrow frequency peak amplitudes was not performed. The response is acceptable. However, the description of the ASCE 4-13 still remains in Section 7.2 of the markup of the revised TR-0916-51502. Since ASCE 4-13 has no relevance with the ISRS provided in TR-0916-51502, remove or minimize the description of ASCE 4-13 to avoid confusion.

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**NuScale Response:**

The references to ASCE 4-13 have been removed from TR-0916-51502.

The following sentences are removed from Section 7.2 of TR-0916-51502:

“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).”

"Reduction of narrow frequency peak fmplitudes was not performed."

The following sentence has been removed from Section 3.1.7 of TR-0916-51502:

“ASCE 4-98 is in the process of being updated to ASCE 4-13 (Reference 10.1.1) and is also referenced for guidance.”

Reference 10.1.1 has been removed from Section 10.1.1 of TR-0916-51502.

**Impact on DCA:**

The technical report TR-0916-51502 Sections 3.1, 7.2, and 10.1 have been revised as described in the response above and as shown in the markup provided with this response.

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.

## 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).

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 equivalent force is in accordance with Section 4.5 of Reference 10.1.7. The ISRS is the

## 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.~~ Not Used.

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.

10.1.3 NUREG-0800, Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants, Section 3.7.3, Seismic Subsystem Analysis Review Responsibilities, Draft Revision 4, September, 2013.

10.1.4 IEEE Standard 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

10.1.5 ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," 2013 Edition with no addenda.

10.1.6 US NRC Regulatory Guide 1.122,"Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," Revision 1, February 1978.

10.1.7 American Society of Civil Engineers, "Seismic Analysis of Safety-Related Nuclear Structures," ASCE 4, 1998.

10.1.8 American Society of Civil Engineers, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," ASCE/SEI 43, 2005.

10.1.9 R.J. Fritz, "The Effect of Liquids on Dynamic Motion of Immersed Solids," Journal of Engineering for Industry, February, 1972.

10.1.10 Meyer, M. et al. "Generalized Barycentric Coordinates on Irregular Polygons". Pages 13-22. Journal of Graphic Tools, Volume 7 Issue 1, November 2002.

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.

NuScale Power™

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

**eRAI No.:** 9310
**Date of RAI Issue:** 04/09/2018

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**NRC Question No.:** 03.09.02-67

In the response of RAI 03.09.02-39, the applicant stated that NuScale is seeking design certification for the NuScale power module (NPM) considering seismic analysis for a single soil type (Soil Type 7) with single time history [certified seismic design response spectra (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 applicant further stated that seismic analysis using a single CSDRS-based time-history and a single soil type input applies only to the NPM model and the seismic design of the reactor building is based on analyses involving multiple time histories and soil types, which includes high frequency CSDRS (CSDRS-HF), as discussed in FSAR Section 3.7.2.4. The staff is unclear whether NuScale is seeking design certification for the NPM at the sites with characteristics consistent with CSDRS inputs only or for both CSDRS and CSDRS-HF inputs. Provide the following information:

1. Is NuScale seeking design certification of components designed for both CSDRS and CSDRS-HF inputs? If the answer is no, add a COL item for the COL applicant to address CSDRS-HF input for component design.
2. Does the NPM seismic analysis using a single CSDRS-based time-history (CSDRS-compatible Capitola) and a Soil Type 7 bound the NPM component seismic stresses for NPM located at hard rocks sites (i.e., CSDRS-HF sites)?
3. If the answer of Subquestion 2 is yes, explain why the RPV support interface loads in the RPV_support_interface_loads.pdf (i.e., a document provided in the eRR during the NPM seismic audit) contain two types of seismic loads, CSDRS and CSDRS-HF (also callled GHFHRRS, Generic High Frequency Hard-Rock Response Spectra). The staff noticed that in some cases, the GHFHRRS interface loads are higher than those of CSDRS.

Include the requested information in the NPM Seismic Report or DCD.

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**NuScale Response:**

The subquestions are addressed individually:

1. For the NPM, NuScale is not seeking design certification for CSDRS-HF inputs. A requirement to address CSDRS-HF input for seismic design of the NPM is included in FSAR Section 3.9. as new COL Item 3.9-12.

2. NuScale has not sought to demonstrate that NPM component seismic stresses based upon the CSDRS compatible input and a soil type 7 always bound NPM component seismic stresses based upon CSDRS-HF compatible input at hard rock sites. As noted in Item 1, the requirement to address CSDRS-HF input at hard rock sites is a COL Item.

3. Because NuScale is not seeking design certification of the NPM for the CSDRS-HF inputs, the CSDRS-HF loads provided in “RPV_support_interface_loads.pdf” are not relevant to the design certification.

COL Item 3.9-12 has been added to FSAR Section 3.9.1 as follows:

A COL applicant that references the NuScale Power Plant design certification will perform a site-specific seismic analysis in accordance with Section 3.7.2.16. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the CSDRS, the standard design of NPM components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demand.

**Impact on DCA:**

The FSAR Tier 2, Section 3.9.1 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 01-61, 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 03.06.03-11, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.08.04-23S1, RAI 03.08.05-14S1, RAI 03.09.02-15, RAI 03.09.02-48, RAI 03.09.02-67, RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-6, RAI 03.09.06-16, RAI 03.09.06-16S1, RAI 03.09.06-27, RAI 03.11-8, RAI 03.11-14, RAI 03.11-14S1, RAI 03.11-18, RAI 03.13-3, RAI 04.02-1S2, RAI 05.02.05-8, RAI 05.04.02.01-13, RAI 05.04.02.01-14, RAI 06.04-1, RAI 09.01.02-4, 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 10.02-3, RAI 10.02.03-1, RAI 10.02.03-2, RAI 10.03.06-1, RAI 10.03.06-5, RAI 10.04.06-1, RAI 10.04.06-2, RAI 10.04.06-3, 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, RAI 14.02-7, RAI 19-31, RAI 19-31S1, RAI 19-38

## Table 1.8-2: Combined License Information Items

| Item No. | Description of COL Information Item | Section |
|---|---|---|
| COL Item 1.1-1: | A COL 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 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 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 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 that references the NuScale Power Plant design certification will list additional site-specific piping and instrumentation diagrams and legends as applicable. | 1.7 |
| COL Item 1.8-1: | A COL 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 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 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 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 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 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 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 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 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~~except Section 2.4.8 and Section 2.4.10. | 2.4 |
| COL Item 2.5-1: | A COL 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 |
| COL Item 3.2-1: | A COL applicant that references the NuScale Power Plant design certification will update Table 3.2-1 to identify the classification of site-specific structures, systems, and components. | 3.2 |

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

| Item No. | Description of COL Information Item | Section |
|---|---|---|
| COL Item 3.9-1: | A COL applicant that references the NuScale Power Plant design certification will provide the applicable test procedures before the start of testing and will submit the test and inspection results from the comprehensive vibration assessment program for the NuScale Power Module, in accordance with Regulatory Guide 1.20. | 3.9 |
| COL Item 3.9-2: | A COL applicant that references the NuScale Power Plant design certification will develop design specifications and design reports in accordance with the requirements outlined under American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III (Reference 3.9-1). A COL applicant will address any known issues through the reactor vessel internals reliability programs (i.e. Comprehensive Vibration Assessment Program, steam generator programs, etc.) in regards to known aging degradation mechanisms such as those addressed in Section 4.5.2.1. | 3.9 |
| COL Item 3.9-3: | A COL applicant that references the NuScale Power Plant design certification will provide a summary of reactor core support structure ASME service level stresses, deformation, and cumulative usage factor values for each component and each operating condition in conformance with ASME Boiler and Pressure Vessel Code Section III Subsection NG. | 3.9 |
| COL Item 3.9-4: | A COL applicant that references the NuScale Power Plant design certification will submit a Preservice Testing program for valves as required by 10 CFR 50.55a. | 3.9 |
| COL Item 3.9-5: | A COL applicant that references the NuScale Power Plant design certification will establish an Inservice Testing program in accordance with ASME OM Code and 10 CFR 50.55a. | 3.9 |
| COL Item 3.9-6: | A COL applicant that references the NuScale Power Plant design certification will identify any site-specific valves, implementation milestones, and the applicable ASME OM Code (and ASME OM Code Cases) for the preservice and inservice testing programs. These programs are to be consistent with the requirements in the latest edition and addenda of the OM Code incorporated by reference in 10 CFR 50.55a in accordance with the time period specified in 10 CFR 50.55a before the scheduled initial fuel load (or the optional ASME Code Cases listed in Regulatory Guide 1.192 incorporated by reference in 10 CFR 50.55a). | 3.9 |
| COL Item 3.9-7: | Not Used. | |
| COL Item 3.9-8: | A COL applicant that references the NuScale Power Plant design certification will develop specific test procedures to allow detection and monitoring of power-operated valve assembly performance sufficient to satisfy periodic verification design basis capability requirements. | 3.9 |
| COL Item 3.9-9: | A COL applicant that references the NuScale Power Plant design certification will develop specific test procedures to allow detection and monitoring of emergency core cooling system valve assembly performance sufficient to satisfy periodic verification of design basis capability requirements. | 3.9 |
| 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 main steam lines, using acoustic resonance screening criteria and additional calculations as necessary (e.g., Strouhal number) to determine if there is a concern. The methodology contained in "NuScale Comprehensive Vibration Assessment Program Technical Report," TR-0716-50439 is acceptable for this purpose. The COL applicant will update Section 3.9.2.1.1.3 to describe the results of this evaluation. | 3.9 |
| COL Item 3.9-11: | A COL applicant that references the NuScale Power Plant design certification will implement a CRDS Operability Assurance Program that meets the requirements described in NUREG-0800, SRP 3.9.4, Revision 3, Acceptance Criteria II.4. | 3.9 |
| COL Item 3.9-12: | A COL applicant that references the NuScale Power Plant design certification will perform a site-specific seismic analysis in accordance with Section 3.7.2.16. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the CSDRS, the standard design of NPM components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demand. | 3.9 |
| COL Item 3.10-1: | A COL applicant that references the NuScale Power Plant design certification will develop and maintain a site-specific seismic and dynamic qualification program. | 3.10 |

## 3.9 Mechanical Systems and Components

### 3.9.1 Special Topics for Mechanical Components

This subsection addresses information concerning methods of analysis for Seismic Category I components and supports, including those designated as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III (Reference 3.9-1), Division 1 Class 1, 2, 3, subsection NG for core support structures, subsection NF for supports, and those not covered by the ASME BPVC as discussed in NUREG 0800 Standard Review Plan (SRP) 3.9.1. Information also is presented concerning design transients for ASME BPVC Class 1 and core support structure components and supports.

RAI 03.09.02-67

COL Item 3.9-12: A COL applicant that references the NuScale Power Plant design certification will perform a site-specific seismic analysis in accordance with Section 3.7.2.16. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the CSDRS, the standard design of NPM components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demand.

The NuScale Power Plant design meets the relevant requirements of the following General Design Criteria (GDC) of 10 CFR 50, Appendix A:

- GDC 1, as it relates to components being designed, fabricated, erected, constructed, tested, and inspected in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety-related functions to be performed. Compliance with GDC 1 is discussed in Section 3.1.
- GDC 2, as it relates to mechanical components of systems being designed to withstand seismic events without loss of capability to perform their safety-related functions. Pursuant to GDC 2, mechanical components are designed to withstand the loads generated by natural phenomena as discussed Section 3.1.1.
- GDC 14, as it relates to the reactor coolant pressure boundary (RCPB) being designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. As discussed below, the design transients and consequent loads and load combination with appropriate ASME code service limits, provide reasonable assurance that the RCPB is designed to maintain the stresses within acceptable limits to accommodate the system pressures and temperatures expected from normal operation including anticipated operational occurrences (AOOs), infrequent events, and accident loading conditions such as safe shutdown earthquake (SSE).
- GDC 15, as it relates to the mechanical components of the reactor coolant system (RCS) being designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The overpressure protection features are designed with sufficient capacity to prevent the RCPB from exceeding 110 percent of design pressure during normal operations and AOOs. Safety-related mechanical components are designed to remain functional under

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

**eRAI No.:** 9310
**Date of RAI Issue:** 04/09/2018

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**NRC Question No.:** 03.09.02-68

In the response to RAI 8911, Question 03.09.02-42, the applicant stated that forces and moments acting on the selected set of nodes from the selected elements on one side of the cross section cut are summed about a point at the centerline of the cross section or interface to obtain the interface resultant forces and moments. The resultant forces and moments are then used for stress analysis of individual components. The applicant is requested to confirm that all the interface loads in the NPM seismic analysis using the selected element approach bound the interface loads considering all elements in the cross section. If not, how this difference is considered in the stress analysis of individual components.

Include the requested information in the NPM Seismic Report.

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**NuScale Response:**

The selected set of nodes and elements for the nodal force and moment summations encompass the entire cross section; i.e. this includes all the elements on one side of the cross section that contain the nodes on the cross section, as required by the ANSYS command FSUM. The nodal force contributions of the selected elements (on one side of the section) are summed and stored. The nodal moment contributions of the selected elements are summed about the geometric center of the nodes that compose the cross section, and stored. The nodes in the geometric center of the cross-sections are named for data retrieval.

**Impact on DCA:**

Technical report TR-0916-51502 Section 8.4.2.4 has been revised as described in the response above and as shown in the markup provided with this response.

| Component Section ID | Name | Elevation, Y (in) | Figure |
|---|---|---|---|
| 17 | {{ | }}2(a),(c) | Figure A-6 |
| 18 | {{ | }}2(a),(c) | Figure A-6 |
| 19 | {{ | }}2(a),(c) | Figure A-7 |
| 20 | {{ | }}2(a),(c) | Figure A-7 |
| 21 | {{ | }}2(a),(c) | Figure A-7 |
| 22 | {{ | }}2(a),(c) | Figure A-7 |

The FSUM command is used by selecting the elements and nodes that compose a full cross section at each component section or interface. The nodal force contributions of the selected elements (on one side of the section) are summed and stored. The nodal moment contributions of the selected elements are summed about the geometric center of the nodes that compose the cross section and stored. The nodes in the geometric center of the cross-sections are named for data retrieval.

#### 8.4.2.5 In-Structure Response Spectra

For each of the locations listed in Table 8-3, the displacements were extracted for each direction in the global Cartesian coordinate system. For response spectrum generation, six different damping ratios were considered: 2, 3, 4, 5, 7 and 10 percent.

ISRS was generated from the seismic model result files, and broadened ISRS with a 15 percent frequency shift calculated for the enveloping ISRS. A typical broadened ISRS is shown in Figure 8-9.

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

**eRAI No.:** 9310
**Date of RAI Issue:** 04/09/2018

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**NRC Question No.:** 03.09.02-72

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. During the audit for NuScale power module seismic analysis, the NRC staff noticed that the following load combination is used in the RPV Service Level D primary stress calculation documented in EC-A011-2278, Rev.0, “Reactor Pressure Vessel Primary Stress Calculation.”

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

The staff requests that the applicant provide a table that shows the type of the EXT load, location of the load, and value of the load in the RPV primary stress calculation.

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**NuScale Response:**

In the RPV Primary Stress Analysis, the EXT term in the Level D load combination includes piping mechanical and thermal loads, support reactions associated with the reactor pressure vessel (RPV) due to deadweight, and reactor vessel internals (RVI) interface loads due to deadweight. Table 1 summarizes the type, location, and value of the applicable piping mechanical and thermal loads. Table 2 summarizes the type, location, and value of the applicable reaction and interface loads.

Table 1 Piping mechanical and thermal loads

| Location[1] | Description | Load type[2] | Load[3] | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | P (lb) | VC (lb) | VL (lb) | MT (in-lb) | ML (in-lb) | MC (in-lb) |
| RPV1-2 | Reactor recirculation valve RRV nozzles | deadweight | {{ | | | | | |
| | | pressure | | | | | | |
| | | thermal | | | | | | }}[2(a),(c)] |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| RPV3-6 | Feedwater nozzles | deadweight | {{ | | | | | |
| | | pressure @ 2,100 psia | | | | | | |
| | | thermal @ 700F | | | | | | |
| RPV7-10 | Main steam nozzles | deadweight | | | | | | |
| | | pressure @ 2,100 psia | | | | | | |
| | | thermal @ 700F | | | | | | |
| RPV11 in | Reactor coolant system RCS injection nozzle inside the vessel | deadweight | | | | | | |
| | | pressure | | | | | | |
| | | thermal | | | | | | |
| RPV11 out and RPV12 | RCS injection nozzle outside the vessel and RCS discharge nozzle | deadweight | | | | | | |
| | | pressure @ 2,100 psia | | | | | | |
| | | thermal @ 700F | | | | | | |
| RPV14-15 in and out | PZR spray nozzles inside and outside the vessel | deadweight | | | | | | |
| | | pressure @ 2,100 psia | | | | | | |
| | | thermal @ 700F | | | | | | |
| RPV16-17 and RPV83 | Reactor ventilation valve RVV nozzles | deadweight | | | | | | |
| | | pressure | | | | | | |
| | | thermal | | | | | | |
| RPV18-19 | Reactor safety valve RSV nozzles | deadweight | | | | | | |
| | | pressure | | | | | | |
| | | thermal | | | | | | |
| RPV20 | High point de-gas nozzle | deadweight | | | | | | |
| | | pressure @ 2,100 psia | | | | | | |
| | | thermal @ 700F | | | | | | |
| RPV23-38 | Control rod drive mechanism CRDM nozzles | deadweight | | | | | | |
| | | pressure | | | | | | |
| | | thermal | | | | | | }}[2(a),(c)] |

1. Pressure and thermal loads are applicable inside the limits of reinforcement at each nozzle location. Piping mechanical and thermal loads are conservatively included outside the limits of reinforcement in the ANSYS analysis.

2. Deadweight loads are due to attached piping/valves and do not include self-weight of the nozzle.

3. Coordinate systems for loads at a given location are documented in the RPV Primary Stress Analysis.

Table 2 Reaction and interface loads

| Location | Load type | Load[1] | | | | | |
|---|---|---|---|---|---|---|---|
| | | Fx (lb) | Fy (lb) | Fz (lb) | Mx (in-lb) | My (in-lb) | Mz (in-lb) |
| RPV support (per support) | deadweight | {{ | | | | | |
| Bottom restraint | deadweight | | | | | | |
| Core support block (per support block) | deadweight | | | | | | |
| Upper core blocks (per block) | deadweight | | | | | | |
| Support cantilever (per cantilever) | deadweight | | | | | | |
| RVI loads at baffle plate (per hanger connection) | deadweight | | | | | | |
| CRDM support (per support pad) | deadweight | | | | | | }}2(a),(c) |

1. Coordinate systems for loads at a given location are documented in the RPV Primary Stress Analysis.

**Impact on DCA:**

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

**Enclosure 3:**

Affidavit of Zackary W. Rad, AF-0618-60343

**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. 410, eRAI No. 9310. 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 June 8, 2018.

Zackary W. Rad