December 18, 2024 Docket No. 52-050

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

SUBJECT: NuScale Power, LLC Responses to NRC Request for Additional Information No. 011 (RAI-10111 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, “Request for Additional Information No. 011 (RAI-10111 R1),” dated November 21, 2023

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

The enclosure to this letter contain NuScale's response to the following RAI question from NRC RAI-10111 R1:

- 3.9.2-1

Enclosure 1 is the proprietary version of the NuScale responses to NRC RAI No. 011 (RAI-10111 R1, Question 3.9.2-1). 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 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirement of 10 CFR § 810. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Elisa Fairbanks at 541-452-7872 or at efairbanks@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 18, 2024.

Sincerely,

Mark W. Shaver

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Prosanta Chowdhury, Senior Project Manager, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10111 R1, Question 3.9.2-1, Proprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10111 R1, Question 3.9.2-1, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-177129

Enclosure 1:

NuScale Response to NRC Request for Additional Information RAI-10111 R1, Question 3.9.2-1, Proprietary Version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10111 R1, Question 3.9.2-1, Nonproprietary Version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10111
Date of RAI Issue: 11/21/2023

NRC Question No.: 3.9.2-1

Regulatory Basis
The NRC regulations in 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion 4, “Environmental and Dynamic Effects Design Bases,” require 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.

Issue
FSAR Appendix 3A.1, “Seismic Analysis,” references TR-121515, “US460 NuScale Power Module Seismic Analysis,” Revision 0, and Appendix 3A.2, “Blowdown Simulation,” references TR-121517, “NuScale Power Module Short-Term Transient Analysis,” Revision 0. TR-121515 describes the methodologies and structural models that are used to analyze the dynamic structural loading of reactor vessel internals (RVI) due to seismic loads acting on the NuScale power module-20 (NPM-20). TR-121517 describes methodologies and models that are used to analyze the transient loads within NPM-20 caused by failure or actuation of piping and valves. NPM-20 might be subjected to transient dynamic loads from various phenomena, including seismic events and transient events such as inadvertent valve openings.

FSAR Section 3.9.1, “Special Topics for Mechanical Components” states that core support structures will meet ASME Boiler and Pressure Vessel Code (BPV Code), Section III service level requirements, and in particular the Service Level C (transient events) and Service Level D (seismic events). These requirements include maximum allowable stresses. NuScale has not provided the ASME BPV Code, Section III, Service Level C or D assessments for RVI and SG tubes and supports for NPM-20. NuScale plans to complete final stress reports during inspection, test, analysis, and acceptance criteria (ITAAC) activities for NPM-20. The NRC staff needs to review the preliminary stress analyses to make a safety finding in the SDAA safety

evaluation for NPM-20.
In NuScale's supplemental information submitted to address Audit Question A-3.9.2-8 item 4 (ML23304A410), NuScale explained the discrepancy between the upper reactor vessel internal (URVI) and lower reactor vessel internal (LRVI) interface modeling and resulting differences between the low-order modes of the double building (DB) structural model and the detailed NPM-20 structural model. {{

}}2(a),(c) The staff cannot agree with this conclusion without additional information, {{

}}2(a),(c) shown in the report and is concerned the modeling assumption may have led to non-conservative seismic loads in that frequency range.

Requested Information
1. During the design certification application (DCA) review for NPM-160, NuScale submitted a detailed response to staff request for additional information (eRAI 8911, Question 09.02- 18 in NuScale transmittal RAIO-0819-66530), which included ASME Service Level D stress assessments. NuScale is requested to provide information at a similar level of detail for seismic and transient loading of NPM-20, including the structural integrity of the control rod assembly guide tubes (CRAGT), in-core instrument guide tubes (ICIGT), and SG tubes and supports. The information should include the following:
• a brief description of the component modelling,
• applied input motion (time history or in-structure response spectrum, coherent incoherent loading at different mounting locations),
• major assumptions,
• acceptance criteria under Service Level C or D condition including deflection limits,
• fluid modelling, mass distribution, damping value(s),

• gap considerations and potential contact with adjacent components and/or walls,
• dominant modes of vibration and frequencies (show images of the mode shapes),
• stress convergence studies,

• stress concentration factors, and
• weld factors (if welds are at or near a limiting location)

2. To support the review of the seismic and transient loading for the SDAA safety evaluation for NPM-20, NuScale is also requested to provide the following information:

• Calculation results for radial deflections of the CRAGTs and ICIGT and assessments of the effects of impacts (if any) between the CRAGTs and ICIGT and the adjacent holes within the various support plates along the height of the riser.
• Quantitatively substantiate the statement that the DB modeling assumption at the URVI/LRVI interface does not lead to non-conservative bias results in the final seismic loading and dynamic response calculations. A comparison between loading calculations performed with and without the assumption corrected for a limiting condition is one option to substantiate NuScale's conclusion.

NuScale Response:

Executive Summary

At the time of the original request, the NuScale Power Module (NPM) representation in the NPM and double building seismic evaluations were different. NuScale has subsequently revised the NPM representation in both seismic evaluations to be identical. The double building seismic evaluation contains a linear representation of the detailed NPM model because the seismic analysis is a harmonic evaluation which does not permit nonlinearities. Whereas, the NPM seismic evaluation is a non-linear time histories evaluation using the same NPM representation and does explicitly include nonlinearites (e.g., containment vessel skirt to reactor building basemat interface). This change alleviates the need to conduct comparison studies because the representation is consistent across both seismic analyses.

Classical engineering Service Level D calculations were performed for reactor vessel internal (RVI) components with loadings generated from the revised NPM seismic evaluation, a material change for the core support block top plate and gussets, an increase in diameter of the socket head cap screws that fasten the top plate to the gussets, an increased diameter of the shoulder stud nut at the core support mounting bracket location, and an increase in diameter of the limiting notched regions of the core support shoulder stud. Each location analyzed met the applicable acceptance criteria for ASME BPVC Section III. {{

}}2(a),(c)

NuScale provides an integrated model of the RVI without the steam generator (SG) for modal and response spectrum analyses. The modal results demonstrate the response of the assembly for postulated dynamic loadings, including seismic and depressurization events. The combined model includes the upper riser, lower riser, and core barrel assemblies. The results of the analysis indicate the most limiting location, other than the in-core instrumentation guide tubes (ICIGTs), is less than {{ }}[2(a),(c)] occurring on the control rod assembly guide tube support plate. This results in a {{ }}[2(a),(c)] margin over the general primary membrane stress intensity allowable Level D stress limit. The ICIGTs experience higher stresses due to their long, unsupported spans, but the stress is below the general primary membrane stress intensity allowable Level D stress limit. All calculated stresses are less than all applicable ASME Service Level D allowable limits.

The RVI representative finite element model (FEM) calculation and the RVI classical engineering calculations demonstrate that the NPM RVI have adequate margin against ASME Service Level D allowable stresses.

A Service Level D FEM analysis was also performed for the SG tubes to demonstrate adequate structural requirements in accordance with ASME BPVC. The scope of this calculation is to calculate the Service Level D stresses for the SG tubes using transient analysis for seismic loads and blowdown loads, including deadweight and pressure loads. The SG ANSYS model includes 1,380 SG tubes, 168 tube supports, 8 backing strips, and 8 lower SG supports. The current qualification scope performed is limited to SG tubes for Level D criteria. Furthermore, to meet the post-accident operability requirement, the more rigorous Level B criteria are used in this qualification. For the SG assembly, a Level B stress limit of 29.37 ksi is used. At the most limiting location, preliminary stress intensity is {{ }}[2(a),(c)], which is under the stress load limit. For shear stresses, the limit is 16 ksi. The analyzed limiting stress is {{ }}[2(a),(c)], significantly under the shear load limit.

Thus for the respective representative FEM models, The RVI representative FEM model and SG both demonstrates adequate margin against code allowable stress under the analyzed ASME BPVC Service Level D transients.

Classical Engineering Calculations

Geometry for Classical Engineering Calculations

The RVI geometry of the components evaluated via classical engineering calculations are displayed in Figure 1, Figure 2, Figure 3, and Figure 4.

Figure 1: Reactor Pressure Vessel Lower Subassembly Model (Reactor Pressure Vessel Shell and Head Not Shown)

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

Figure 2: Core Support Mounting Components

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Figure 3: Lower Riser Assembly Components

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

Figure 4: Core Barrel Assembly Components

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

Methodology for Classical Engineering Calculations

Classical engineering calculations refer to hand calculations performed using closed-form equations for components listed in Table 1 for ASME Code qualification to Service Level D conditions and load combinations.

Table 1: Reactor Vessel Internals Classical Calculations Evaluated Components

Component	Material
Core Barrel	SA-965, Grade F304, 0.03 percent Max Carbon
Core Support Mounting Bracket Gusset (A & B)	SA-182, Grade FXM-19
Core Support Mounting Bracket Shoulder Nut	SA-479, Type 316, 0.03 percent Max Carbon, Strain Hardened
Core Support Mounting Bracket Alignment Pin	SA-479, Type 304
Core Support Mounting Bracket Shoulder Stud	SA-479, Type 316, 0.03 percent Max Carbon, Strain Hardened
Core Support Mounting Bracket Threaded Insert	SA-479, UNS S21800
Core Support Mounting Bracket Top Plate	SA-182, Grade FXM-19
Hex Head Cap Screw	SA-479, UNS S21800
Lower Core Plate – Tabs Only	SA-240, Type 304, 0.03 percent Max Carbon
Lower Riser Section	SA-240, Type 304, 0.03 percent Max Carbon
Socket Head Cap Screw 1-8UNC-2A X 3.500	SA-479, Type 304, 0.03 percent Max Carbon
Socket Head Cap Screw 1½-12UNF-2A X 4.500	SA-479, Type FXM-19
Threaded Insert	SA-479, UNS S21800
Upper Core Plate Attachment Block	SA-240, Type 304, 0.03 percent Max Carbon
Upper Core Plate – Tabs Only	SA-240, Type 304, 0.03 percent Max Carbon
Upper Support Block	SA-479, Type 304, 0.03 percent Max Carbon
Upper Support Block Socket Head Cap Screw 1.750-8UN-2A X 8.500	SA-479, Type 316, 0.03 percent Max Carbon, Strain Hardened

Calculations are performed on specific joints and components. A guide to the specific calculations performed for comparison to ASME criteria is provide in Table 2.

Table 2: Classical Engineering Specific Geometries

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

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

The reactor pressure vessel depressurization plus safe shutdown earthquake (SSE) plant event load combination bounds every other Service Level D load combination and is utilized in this calculation. The load combination is provided in Table 3.

Table 3: Reactor Vessel Internals Service Level D Load Combination

Plant Event	Service Level	Load Combinations	Allowable Limit
RPV Depressurization + SSE	D	PD + DW + EXT + SCR + SRSS[(1)][SSE + DFL, MAX(CVCS, ECCS, RSV)]	D
Note(s): (1) Square root of the sum of squares			

Table 3 load definitions:

- CVCS (chemical and volume control system): Loads from chemical and volume control system injection line break. This is categorized as a design basis pipe break load.
- DFL (dynamic fluid load): Water hammer loads are not currently available and are therefore excluded from this qualification.

- DW (deadweight): Loads determined by applying a 1g acceleration in the vertical direction in ANSYS. These loads do not apply to all fastener calculations.
- ECCS (emergency core cooling system): This includes spurious valve actuation of reactor vent valves and reactor recirculation valves, neither of which apply in the scope in of this qualification.
- EXT (mechanical loads such as RVI support reactions, fuel assembly weight, etc.): This includes bellows expansion, fuel assembly hold down spring force, and fastener preload.
- PD (operating pressure difference): The operating pressure difference across the RVI does not vary significantly between power levels, and therefore PD is neglected.
- RSV (reactor safety valve): Loads are not applicable to the in-scope components of this analysis.
- SCR (SCRAM loads): Sudden shut down of reactor by operator or “reactor trip”
- SSE (safe shutdown earthquake): in-structure response spectra (ISRS) or time histories.

The forces and moments utilized in the RVI classical engineering calculation are generated by post-processing the seismic and blowdown runs for each module location and for every time point. The maximum values for forces (Fx, Fy, and Fz) and moments (Mx, My, and Mz) are determined independent of time, module location, and load condition (e.g., the maximum Fx force could be from one module for one seismic run and the maximum Fy force could be from another module for a different seismic run, etc.).

Key Assumptions for Classical Engineering Calculations

There are three key assumptions are used in the evaluation.

- Preliminary loading report for RVI is utilized as the source for non-SSE loadings
- Geometry discrepancies are documented for an update and verification
- Planned design changes are incorporated into the classical engineering calculations
    - The core support mounting bracket top plates and gussets will be made of SA-182 Grade FXM-19 in lieu of SA-182 Grade F304 and SA-240 Type 304, respectively.
    - Socket head cap Screws of size 1¼-12UNF-2A X 4.500 will be changed to 1½ -12UNF-2A X 4.500.
    - The diameter of the core support mounting bracket shoulder nuts will be increased from 5.375” to 5.500”.

- The non-threaded portions of the core support mounting bracket shoulder studs will have notches decreased so that the limiting diameter is 2.875” in lieu of 2.688”.

The ASME design report and data package required by ASME Code Section III NCA-3550 provides the evidence assumptions are verified. The ITAAC closure of 02.01.01 and 02.01.02 documents the verification and closure.

Materials for Classical Engineering Calculations

The RVI assembly Level D temperature and pressure conditions are 540 degrees F and 2000 psi, respectively. Table 4 lists the material properties for the components at 540 degrees F.

Table 4: Material Properties at 540 Degrees F

Material	Design Stress Intensity, S_m (ksi)	Yield Strength, S_y (ksi)	Ultimate Tensile Strength, S_u (ksi)	Modulus of Elasticity ($x10^6$), E (psi)	Thermal Expansion Coefficient, α ($x10^{-6}$) (1/°F)
SA-182 Grade FXM-19(1)	29500	38240	88540	25.7	9.10
SA-240, Type 304	17140	19000	63400	25.7	9.78
SA-479, Type 304	17140	19000	63400	25.7	9.78
SA-479, Type 316	25700	47980	77100	25.7	9.78
SA-479, Type XM-19	29500	38240	88540	25.7	9.10
SA-479, UNS S21800	18080	27120	78740	23.4	9.78
SA-965, Grade F304	17140	19000	59200	25.7	9.78

Results

Table 5 shows the summary of results for the most limiting locations analyzed with the classical engineering calculations. Each location analyzed met the applicable acceptance criteria for ASME BPVC Section III. {{ }}2(a),(c) Table 6 lists the applicable ASME BPVC Section III Criteria for the 2017 Edition of the Code.

Two welds are in scope of the classical engineering analysis. The lower riser to upper core plate weld and the core barrel to lower core plate weld. Both welds have a 0.5 weld quality factor applied to the general membrane stress intensity and shear stress limits.

Table 5: Reactor Vessel Internal Component Level D Summary of Results

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

Table 6: Classical Calculations Level D Acceptance Criteria

Component Type	Stress Category	Limit	ASME Code Reference
All Except Thread Structural Fasteners	General primary membrane stress, P_m	Min [2.4(S_m) and 0.7(S_u)]	NG-3225, XXVII-3210
	General primary membrane plus bending stress, $P_m + P_b$	1.5(P_m Limit)	NG-3225, XXVII-3230
	Pure shear	2(0.6S_m)	NG-3225(a), NG-3227.2
	Bearing	2(S_y)	NG-3225, NG-3227.1(a)
Threaded Structural Fasteners	Tensile Stress	Min[2.4(S_m) and 0.7(S_u)]	NG-3235(b), XXVII-3210
	Shear Stress	0.42(S_u)	NG-3235(b), XXVII-3520
	Membrane plus Bending Stress	1.5(P_m Limit)	XXVII-3230

The acronyms used in Table 6 are as follows:

- S_m: design stress intensity
- S_y: yield strength
- S_u: ultimate tensile strength
- P_m: general primary membrane stress intensity
- P_b: bending stress

Changes implemented in the classical calculations:

- Seismic loads are updated to the reflect the most up to date inputs
- The core support mounting bracket top plate material changed from SA-182, Grade F304, 0.03 percent max carbon steel to SA-182, grade FXM-19 steel
- Core support mounting bracket gusset (A and B) material changed from SA-240, Type 304, 0.03 percent Max Carbon to SA-182, Grade FXM-19
- RPV Depressurization + SSE Service Level D load combination updated to be consistent with FSAR Table 3.9-7a, reflected in SDAA Revision 2
- Seismic loads are consistent with TR-121515, Revision 1, "US460 NuScale Power Module Seismic Analysis".
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}}2(a),(c)

- {{

}}2(a),(c)

Reactor Vessel Internals Bounding Finite Element Model Calculation Methodology

The ANSYS project developed for the RVI bounding calculation combines three finite element models (Upper RVI, Lower RVI, and Core Region) for the purposes of seismic and blowdown evaluation. The three models are combined using multiple ANSYS capabilities (e.g., joints, constraint equations, contact, etc.). Natural frequencies of the RVI assembly are computed via a modal evaluation. Stresses are generated for applicable ASME Service Level D load combinations via static structural and multi-point response spectra analyses. The components are modeled using current engineering drawings with simplifications to remove small components and openings (screws and open slots), simplified tubes modeled as hollow beams, support plates modeled as shells, and the bellows modeled as springs and point masses. The models implement stainless steel 304 for elastic material properties and densities, evaluated at 550 degrees F. The FEMs generated from these geometries and properties provide the analytical basis for seismic inputs.

Modal analysis of the combined RVI model is performed in ANSYS. The analysis is configured to search for modes between 0 – 100 Hz, with a maximum of 1500 modes. Fixed boundary conditions in all degrees-of-freedom are applied to the upper RVI (URVI) to RPV baffle plate interface at the URVI hanger plate and the core region at the core support mounting blocks (CSMB) to lower RPV interface. Fixed boundary conditions in the lateral degrees-of-freedom are applied to the URVI at the steam generator support to URVI set screw locations.

The cumulative mass fraction in the X- and Y-directions for frequencies up to 100 Hz is approximately {{ }}[2(a),(c)], and in the Z-direction is approximately {{ }}[2(a),(c)]. The cumulative effective mass ratio as a function of frequency for the combined RVI assembly is plotted in Figure 5. The X- and Y-Direction cumulative mass fractions follow closely due to the general axisymmetric nature of the RVI, while the Z-axis participation diverges from the horizontal modes with primary participation above {{ }}[2(a),(c)].

In-structure response spectra are generated and applied at the CRDM ends, CSMBs, hanger plate, and six set screw locations along the height of the upper riser shell. The ISRS are calculated from acceleration time histories for seismic and RPV depressurization loadings, and utilize four percent damping. The square-root-sum-of-the-squares (SRSS) method is used to combine modal responses.

Figure 5: Effective Mass Ratio Across Frequency Range

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}}2(a),(c)

The resultant stresses calculated in the response spectrum analyses are combined for the seismic and blowdown loadings and the maximum stress intensity in the RVI assemblies (excluding the ICIGTs) is calculated to be {{ }}2(a),(c) in the URVI lower control rod assembly guide tube support plate as shown in Figure 6.

Figure 6: Combined Maximum Stress Intensity for Reactor Vessel Internals

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

The resultant stresses calculated in the response spectrum analyses are combined for the seismic and blowdown loadings and the maximum stress intensity in the RVI ICIGTs is calculated to be {{ }}2(a),(c) as shown in Figure 7.

Figure 7: Maximum Combined (Left) and Static Structural (Right) In-Core Instrumentation Guide Tubes Stress Intensity

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

The stress intensity results displayed in Figure 6 and Figure 7 correlate with ASME Code total stresses, where total stresses have three elements: membrane stress, bending stress, and peak stress. Mandatory Appendix XXVII of Section III of the ASME Code defines the criteria for ASME Service Level D. Subarticle XXVII-3200 defines the elastic analysis criteria for ASME Service Level D. There are general primary membrane stress criteria, local primary membrane stress criteria (it should be noted, this criteria is not applicable to Subsection NG components, as there is no local primary membrane stress criteria applicable to NG components), and a membrane plus bending criteria. The general primary membrane stress intensity and the general primary membrane plus bending stress intensity at 550 degrees F for these evaluations are 40.92 ksi and 61.38 ksi, respectively. The maximum stress intensity for any RVI component subjected to ASME Service Level D loading combinations is below the general primary membrane stress intensity limit.

The combined RVI stress results demonstrate considerable margin at each analyzed RVI location with significant margin.

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}}2(a),(c)

Steam Generator Tubes Finite Element Model Calculation Methodology

The ANSYS model developed for the SG assembly includes 1,380 SG tubes, 168 tube supports, 8 backing strips, and 8 lower SG support based on engineering drawings. The pressures and temperatures used in the analysis are described in Table 7, and the materials are described in Table 8.

Table 7: Pressure and Temperature Parameters

	Parameter	Condition	Value	Unit
Primary side	Pressure	Normal operating	2000	psia
	Temperature	100 percent RTP	540	°F
Secondary side	Pressure	full power	475	psia
	Temperature	full power	250	°F

Table 8: Materials for Analysis

Component	Material
Tube	SB-163, UNS N06690
Tube Support	SA-240, TYPE 304
Backing Strip	SA-240, TYPE 304
Lower SG Support	SA-240, TYPE 304

The load combinations for ASME Service Level D are defined in FSAR Table 3.9-3. Within these loads, the bolt load is not applicable to the SG. Piping thermal expansion load (M) is negligible because by comparing RPV and SG tubes, the RPV experiences higher temperature, but its thermal expansion coefficient is lower so the expansion of RPV and SG tubes are similar. Therefore, the M load is not included in this evaluation. Dynamic fluid loads (DFL) are mainly on DHRS, activated from a seismic event. Since, the loads originate from a location distant from the SG tubes, the DFL on SG is deemed insignificant and not included in this evaluation. Depressurization loads (CVCS, ECCS, or RSV) are bounded by blowdown loads (LOCA). Based on the load combinations listed in FSAR Table 3.9-3, the most limiting event is “RPV Depressurization + SSE” because this event includes more loads than other events.

The pressure load uses the normal operating pressure of 2000 psi difference. No additional deadweight is added, other than the components in the analysis, and a differential temperature gradient in discrete bins represents the changes in density. Due to the extensive variants of load for seismic response frequencies and soil selections, the two most limiting cases are

represented at {{
}}2(a,c). For LOCA loads, the the most limiting load is both reactor vent valves opening, therefore, is used.

Figure 8 shows the combined FEM for the SG on which these loads were applied.

Figure 8: Combined Steam Generator Finite Element Model {{

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

The resultant loads were compared against the allowable ASME Service Level B limits as shown in Table 9.

Table 9: Allowable Stress Limits for Service Level B

Stress Category	Code Paragraph	Criterion	Allowable
General Primary Membrane Stress Intensity, P_m	Table XIII-3110-1	1.1 x S_m	1.1*(26.7) = 29.37 ksi
Average Primary Pure Shear Stress	XIII-3720	0.6 x S_m	0.6*(26.7) = 16.02 ksi

The most limiting location is near the bottom of the tubes, where the most stresses are seen. The total stress intensity is {{ }}2(a),(c) , which is under the allowable for general primary membrane stress intensity of 29.37 ksi for ASME Service Level B criteria. For shear stresses, the limiting stress {{ }}2(a),(c) is lower than the allowable 16 ksi, thus being significantly under the allowable limit for ASME Service Level B criteria. The associated allowable limits for ASME Service Level D are more than double the ASME Service Level B allowable limits (e.g., ASME Service Level D general primary membrane stress intensity, P_m, is the lesser (2.4 x S_m, 0.7 x S_u) per Subsection XXVII-3210). In addition, limiting total stress intensity value includes membrane, bending, and peak stresses; therefore, the stress intensity compared to the general primary membrane limit is conservatively high.

The shortest and longest tubes in the SG were analyzed to compare to the Comprehensive Vibration Assessment Program Model and pre-test predictions for TF-3 testing. {{

}}2(a),(c)

Conclusions

NuScale's comprehensive FEM analysis and classical engineering calculations of the RVI assemblies for ASME BPVC Service Level D stress conditions demonstrates the analyzed locations pass the ASME stress criteria, and therefore are considered sufficient for the functional application.

NuScale's comprehensive preliminary FEM analysis SG tube and RVI assemblies for ASME BPVC Service Level D stress conditions demonstrates each location passes the ASME stress criteria.

The following materials were made available for audit in support of RAI 3.9.2-1:

- EC-107251, Revision 2, "NPM-20 Seismic Simulation"
- EC-157339, Revision 1, "ASME Code Level D Evaluation of the Steam Generator Tubes"
- EC-157683, Revision 1, "ASME Code Qualification for Service Level D Condition of RVI Components - Classical Engineering Calculations"
- EC-170084, Revision 1, "ASME Service Level D Finite Element Evaluation of the Reactor Vessel Internals in Support of RAI-10111"
- EC-170090, Revision 0, "NPM-20 Seismic Model Generation"
- EC-172428, Revision 0, "US460 Double Building Model with Detailed Representation of NuScale Power Module"

Impact on US460 SDAA:

There are no impacts to US460 SDAA as a result of this response.

Enclosure 3:

Affidavit of Mark W. Shaver, AF-177129

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of 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 response by which NuScale develops its NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 10111 R1, Question 3.9.2-1) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response 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 RAI 10111 R1, Question 3.9.2-1. The enclosures contain 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 December 18, 2024.

Mark W. Shaver

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