January 21, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 136 (eRAI No. 8933) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 136 (eRAI No. 8933)," dated August 05, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 136 (eRAI No.8933)," dated October 31, 2018

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

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Question from NRC eRAI No. 8933:

- 03.07.02-16

This letter and the enclosed response 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

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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8933

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8933

NUSCALE
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Response to Request for Additional Information

eRAI No.: 8933
Date of RAI Issue: 08/05/2017

NRC Question No.: 03.07.02-16

10 CFR 50 Appendix S requires that the safety functions of structures, systems, and components (SSCs) must be assured during and after the vibratory ground motion associated with the Safe Shutdown Earthquake (SSE) through design, testing, or qualification methods.

In FSAR Section 3.7.2.1.2.1, the staff noted that the dry dock is assumed to be full of water and part of the UHS in the seismic analysis. The nominal water level is at EL. 94 ft. In FSAR Section 9.1.3, the staff also noted that the dry dock can be drained partially or completely to support plant operations. In FSAR Section 9.1.3.3.5, the staff further noted that a failure of the dry dock gate while the dry dock is empty could result in a decrease in water level at the UHS pool by about 12 ft. Since the dry dock contains a large body of water, draining of a large mass of water could affect the dynamic characteristics of the SASSI and ANSYS models thereby potentially affecting the seismic demand based on full dry dock assumption. Therefore the applicant is requested to provide a technical basis for not considering different water level conditions for the dry dock in the seismic analysis. In addition, the applicant should address the effect of potential variation in water level of the UHS on the seismic analysis of the Reactor Building (RXB) and NuScale Power Module (NPM) including the analyses conducted in FSAR 3.7.2.9.1 to address the effect of operation with less than the full complements of NPMs. The applicant should also describe in the FSAR the analysis and design criteria to ensure that no adverse seismic interaction occurs between the dry dock gate and adjacent Seismic Category I SSCs.

NuScale Response:

The NRC provided four feedback statements regarding the NuScale response to RAI 8933, Question 3.07.02-16, dated October 31, 2018. A supplement to that response is provided herein to address the NRC statements.

NRC Statement

1. The applicant did not fully address the items listed in the closure plan.

   a) For example, there was no information demonstrating that the dry dock gate has been designed to the bounding load case and will not fail under an SSE as described in the Closure Plan.

Response

The results from investigating the effects of seismic responses on the structures, systems, and components adjacent to the dry dock gate (DDG) were not addressed in the initial response to the request for additional information. The NuScale calculation that provides the DDG structural analysis was reviewed during the audit of Final Safety Analysis (FSAR) Sections 3.7 and 3.8 during the week of December 3, 2018, and found to be acceptable at that time.

The DDG separates the dry dock from the reactor pool in the Reactor Building (RXB). The DDG is located near column line RX-3 and spans approximately 32 ft in the north-south direction. Details of the DDG are shown in a NuScale drawing that was also reviewed during the audit. When closed, the DDG contacts the pool liner on three sides to seal tightly. When not in use, the DDG is designed to rotate and rest against the north pool wall along column line RX-4. The DDG is supported by hinges on the north wall and operated by a linear actuator. It was analyzed for the worst-case scenario of an empty dry dock with seismic and hydrodynamic loading. That calculation provided the basis for sizing the main DDG components (i.e., horizontal beams, skin plates, and hinges mounted on the north wall) and calculated the wave height in the reactor pool for the projected area of the DDG.

The design requirements of the DDG are as follows:

- Design for the heaviest expected load or combination of loads
- Design to support full water height on the side opposite the dry dock area while the dry dock is empty
- Water tight
- Provide adequate freeboard to prevent water sloshing over the DDG
- Meet Seismic Category II (SC-II) requirements
    - The safe shutdown earthquake (SSE) is used for evaluating SC-I and SC-II structures. The SSE is defined as the loads generated by the SSE specified for

the plant, including the associated hydrodynamic loads and dynamic incremental soil pressure.

- Meet the following codes and standards
  - ANSI/AISC N690-12, “Specification for Safety-Related Steel Structures for Nuclear Facilities”
  - ASCE 4-98, “Seismic Design of Safety-Related Nuclear Structures”
  - ACI 349-06, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary”
  - AISC 360-10, "Specification for Structural Steel Buildings"

The DDG is evaluated by performing static structural analyses for the specified loads. Acceptance criteria are specified by AISC N690-12 and AISC 360-10. The AISC codes consider each loading type (e.g., tension, flexure, shear) separately. While the individual components can be extracted from the finite element model, the combined stress may be more efficiently evaluated. The allowable tension and flexure stresses are the yield strength divided by 1.67 (see AISC 360-10, Chapters D and F). The allowable shear stress uses the same factor of safety, but reduces the nominal strength to 60 percent of yield (AISC 360-10, Chapter G). This is the same factor used in relating shear-to-tensile failure in the von Mises failure criterion. For non-hollow structural section members subject to torsion (shear) and combined stress, the normal stress and shear stress are evaluated independently, using the same allowable stresses listed above (see AISC 360-10, Paragraph H3.3). Because the von Mises stress includes both shear and normal loading, it is acceptable to compare the von Mises equivalent stress to the AISC-allowable tensile stress for evaluating the gross area. Some constrained local yielding is permitted adjacent to areas that remain elastic (AISC 360-10, Paragraph H3.3). In some cases, the areas exceeding the allowable tensile stress are large enough to merit evaluating the component stresses individually.

These evaluations and calculations demonstrate that the DDG has been designed to the bounding load case and will not fail under a safe shutdown earthquake, as described in the NuScale closure plan.

NRC Statement

b) No technical basis has been provided in the response for not considering an uncracked building model and the CSDRS-HF in the analysis with the empty dry dock. Consideration of only the NPM stiffness variation may not adequately reflect the potential impact of the building stiffness variation on the RXB seismic demand when the dry dock

is empty. The applicant is requested to provide the technical basis for selecting the parameters of the empty dry dock model.

Response

The empty dry dock analysis was run for the following parameters:

- Three RXB models with an empty dry dock for three NuScale Power Module (NPM) stiffness cases
    - Model 1: RXB model with nominal NPM stiffnesses
    - Model 2: RXB model with NPM stiffnesses multiplied by 1.3 to create an approximately +15% NPM frequency change in dominant modes
    - Model 3: RXB model with NPM stiffnesses divided by 1.3 to create an approximately -15% NPM frequency change in dominant modes
- 4 percent damping ratio for ISRS generation and lug support reaction calculation
- 7 percent damping ratio for force and moment calculation
- Cracked-concrete condition
- Soil type 7
- Capitola CSDRS-compatible seismic input

The intent of this sensitivity study was to assess the structure for the most limiting case and compare that to the design basis. In so doing, the bullets above tend to be the controlling parameters for the design of the structure and inputs to subsystem analysis. Based on analysis results, the cracked condition with CSDRS-compatible Capitola time history is the governing case. This was observed from numerous sets of previously performed runs. In addition, this set of inputs was used for other sensitivity studies, and, for consistency, the same parameters were used for this work. The uncracked condition and CSDRS-HF results were not presented for the empty dry dock study because they are not the controlling parameters.

NRC Statement

2. The applicant is requested to provide comparisons of the seismic demand (forces and moments) at the NPM support locations (lugs and the skirt) for the corresponding analysis cases with both the empty and full dry dock models.

Response

The following tables compare the results from the empty and full dry dock models. The maximum reactions at the NPM lug and containment vessel (CNV) base skirt supports are compared between the RXB models with a full dry dock and with an empty dry dock. The lug reactions are obtained from the beam elements connected to the NPM lug. The CNV skirt reactions are obtained from the skirt springs. The NPM lug and base skirt support reactions for the full and empty dry dock conditions are provided below. These tables also show the results from the Reactor Building FSAR analyses. The comparison of NPM lug and skirt reactions from the full and empty dry dock conditions shows the reactions are similar for both conditions.

Table 1. Comparison of Lug Reactions with FSAR Results (Capitola Input and Nominal NPM Stiffness)

Dry Dock Condition	West Wing Wall N-S Lug Reaction (kips)	Pool Wall E-W Lug Reaction (kips)	East Wing Wall N-S Lug Reaction (kips)
Full	1,333	1,392	1,377
Empty	1,319	1,273	1,277

Table 2. Comparison of Maximum Forces in NPM Skirt Supports with FSAR Results (Capitola Input and Nominal NPM Stiffness)

Dry Dock Condition	CNV Skirt E-W Reaction (kips)	CNV Skirt N-S Reaction (kips)	CNV Skirt Vertical Reaction (kips)
Full	524	455	1,625
Empty	539	452	1,645

NRC Statement

3. The sensitivity study was performed for one concrete condition (cracked) with Soil Type 7 and one CSDRS-compatible Capitola input. The ISRS from the sensitivity study for the empty dry dock model were then compared against the enveloped RXB ISRS (e.g., FSAR Figure 3.7.2-107). However, this comparison may not be appropriate for the NPM system because the seismic input for the NPM analysis is not based on the above enveloped RXB ISRS. Therefore

the applicant is requested to provide comparisons that is consistent with the NPM support input used for the detailed NPM analysis.

Response

The seismic inputs used for the NPM are time histories, and do not lend themselves to direct comparison from one case to another. Instead, direct comparisons between the empty and full dry dock condition are shown in the response to NRC Statement 2, based on consistent input parameters for the RXB model in lieu of comparisons on an NPM analysis basis. In addition, direct comparisons of floor ISRS at El. 50', El. 75', El.100' and ISRS at a selected Reactor Building crane wheel node, 29545, due to the Capitola seismic input are provided below. The X (East-West), Y (North-South), and Z (vertical) direction ISRS for the floors and at the crane wheel node are shown in FSAR Figure 3.7.2-156 through Figure 3.7.2-159 in the attached FSAR Change Package. The comparison shows that the ISRS are not adversely affected by an empty dry dock.

NRC Statement

4. No FSAR markup has been provided with the response. The applicant is requested to include in the appropriate FSAR sections a summary of the sensitivity studies performed for the empty dry dock cases for the DCD. The applicant should also provide a COL item for the COL applicant to demonstrate that for the specific site conditions, the site-specific demand is bounded by the DCD capacity for an empty dry dock condition.

Response

Final Safety Analysis Report, Tier 2, Section 3.7.2 and Table 1.8-2 have been revised as shown in the attached.

RAI 01-61, RAI 02.04.13-1, RAI 03.04.01-4, RAI 03.04.02-1, RAI 03.04.02-2, RAI 03.04.02-3, RAI 03.05.01.03-1, RAI 03.05.01.04-1, RAI 03.05.02-2, RAI 03.05.03-4, RAI 03.06.02-6, RAI 03.06.02-15, RAI 03.06.03-11, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-6S1, RAI 03.07.02-6S2, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.07.02-15S5, RAI 03.07.02-16S1, RAI 03.07.02-23S1, RAI 03.07.02-26, RAI 03.08.04-3S2, RAI 03.08.04-23S1, RAI 03.08.04-23S2, RAI 03.08.04-23S3, RAI 03.08.05-14S1, RAI 03.09.02-15, RAI 03.09.02-48, RAI 03.09.02-67, RAI 03.09.02-69, 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 05.02.03-19, RAI 05.02.05-8, RAI 05.04.02.01-13, RAI 05.04.02.01-14, RAI 05.04.02.01-19, RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-19, RAI 06.02.06-22, RAI 06.02.06-23, RAI 06.04-1, RAI 09.01.01-20, 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 11.01-2, RAI 12.03-55S1, 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, RAI 20.01-13

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, except Section 2.4.8 and Section 2.4.10.	2.4

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

Item No.	Description of COL Information Item	Section
COL Item 3.7-10:	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific configuration analysis that includes the Reactor Building with applicable configuration layout of the desired NuScale Power Modules. The COL applicant will confirm the following are bounded by the corresponding design certified seismic demands: 1) The in-structure response spectra of the standard design at the foundation and roof. See FSAR Figure 3.7.2-107 and Figure 3.7.2-108 for foundation in-structure response spectra and Figure 3.7.2-113 for roof in-structure response spectra. 2) The maximum forces in the NuScale Power Module lug restraints and skirts. See Table 3B-28. 3) The site-specific in-structure response spectra for the NuScale Power Module at the skirt support will be shown to be bounded by the in-structure response spectra in Figure 3.7.2-156 and Figure 3.7.2-157. The site-specific in-structure response spectra for the NuScale Power Module at the lug restraints will be shown to be bounded by the in-structure response spectra in Figure 3.7.2-158 through Figure 3.7.2-163. 4) The maximum forces and moments in the ~~east and~~ west wing wall~~s~~ and pool wall~~s~~. See ~~FSAR Table 3.7.2-32~~Table 3B-22b and Table 3B-23b. 5) Not Used.~~The site-specific in-structure response spectra for the fuel storage racks will be shown to be bounded by the in-structure response spectra in Figure 3-6 through Figure 3-14 of TR-0816-49833.~~ 6) The site-specific in-structure response spectra shown immediately below will be shown to be bounded by their corresponding certified in-structure response spectra: • Reactor Building north exterior wall at EL 75′-0″: bounded by in-structure response spectra in Figure 3.7.2-110 • Reactor Building west exterior wall at EL 126′-0″: bounded by in-structure response spectra in Figure 3.7.2-112 • Reactor Building crane wheels at EL 145′-6″: bounded by in-structure response spectra in Figure 3.7.2-114 • Control Building east wall at EL 76′-6″: bounded by in-structure response spectra in Figure 3.7.2-119a and Figure 3.7.2-119b • Control Building south wall at EL 120′-0″: bounded by in-structure response spectra in Figure 3.7.2-121a and Figure 3.7.2-121b If not, the standard design will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands.	3.7
COL Item 3.7-11:	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific analysis that~~, if applicable,~~ assesses the effects of soil separation. The COL applicant will confirm that the in-structure response spectra in the soil separation cases are bounded by the in-structure response spectra shown in FSAR Figure 3.7.2-107 through Figure 3.7.2-122.	3.7
COL Item 3.7-12:	A COL applicant that references the NuScale Power Plant design certification will perform an analysis that uses site-specific soil and time histories to confirm the adequacy of the fluid-structure interaction correction factor.	3.7
COL Item 3.7-13:	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific analysis that assesses the effects of non-vertically propagating seismic waves on the free-field ground motions and seismic responses of seismic Category I structures, systems, and components.	3.7
COL Item 3.7-14:	A COL applicant that references the NuScale Power Plant design certification will demonstrate that the site-specific seismic demand is bounded by the FSAR capacity for an empty dry dock condition.	3.7
COL Item 3.8-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific program for monitoring and maintenance of the Seismic Category I structures in accordance with the requirements of 10 CFR 50.65 as discussed in Regulatory Guide 1.160. Monitoring is to include below grade walls, groundwater chemistry if needed, base settlements and differential displacements.	3.8
COL Item 3.8-2:	A COL applicant that references the NuScale Power Plant design certification will confirm that the site-~~-~~independent Reactor Building and Control Building are acceptable for use at the designated site.	3.8

RAI 03.07.02-16S1

3.7.2.1.2 Effect of an Empty Dry Dock

RAI 03.07.02-16S1

A study was performed to determine the effect of an empty dry dock on the response of the RXB. Three separate SASSI models were created for this purpose. The first was the RXB modeled with nominal NPM stiffnesses. The second was an RXB model with NPM stiffnesses multiplied by 1.3, resulting in an approximate +15 percent NPM frequency change in dominant modes. The third model included NPM stiffnesses divided by 1.3, resulting in an approximate -15 percent NPM frequency change in dominant modes. The following parameters were also used in the study:

RAI 03.07.02-16S1

- One set of CSDRS-compatible seismic inputs: Capitola.

RAI 03.07.02-16S1

- One soil type: Soil Type 7.

RAI 03.07.02-16S1

- One concrete condition: cracked.

RAI 03.07.02-16S1

- Two structural concrete damping ratios: 4 percent for ISRS generation and lug support reaction calculation and 7 percent structural damping for force and moment calculation.

RAI 03.07.02-16S1

The maximum forces and moments in the four RXB exterior walls and in the four walls around the dry dock, the lug support reactions at the 12 NPMs, and forces and moments in one pilaster in the north wall at column line RX-4, were calculated for the empty dry dock condition and compared with the corresponding design capacities based on the full dry dock condition. See Table 3.7.2-59 and Table 3.7.2-60 for a sample of results.

RAI 03.07.02-16S1

Comparisons between floor ISRS and ISRS at the Reactor Building crane wheels were also made. These plots can be found in Figure 3.7.2-172 through Figure 3.7.2-175.

RAI 03.07.02-16S1

Based on the comparison of the seismic demands and design capacities, the empty dry dock condition is bounded by the RXB design, which is based on the full dry dock condition.

RAI 03.07.02-16S1

COL Item 3.7-14: A COL applicant that references the NuScale Power Plant design certification will demonstrate that the site-specific seismic demand is bounded by the FSAR capacity for an empty dry dock condition.

RAI 03.07.02-16S1

Table 3.7.2-59: Comparison of Empty Dry Dock Condition Lug Reactions with Final Safety Analysis Report Results (Capitola Input and Nominal NuScale Power Module Stiffness)

Dry Dock Condition	West Wing Wall N-S Lug Reaction (kips)	Pool Wall E-W Lug Reaction (kips)	East Wing Wall N-S Lug Reaction (kips)
Full	1,333	1,392	1,377
Empty	1,319	1,273	1,277

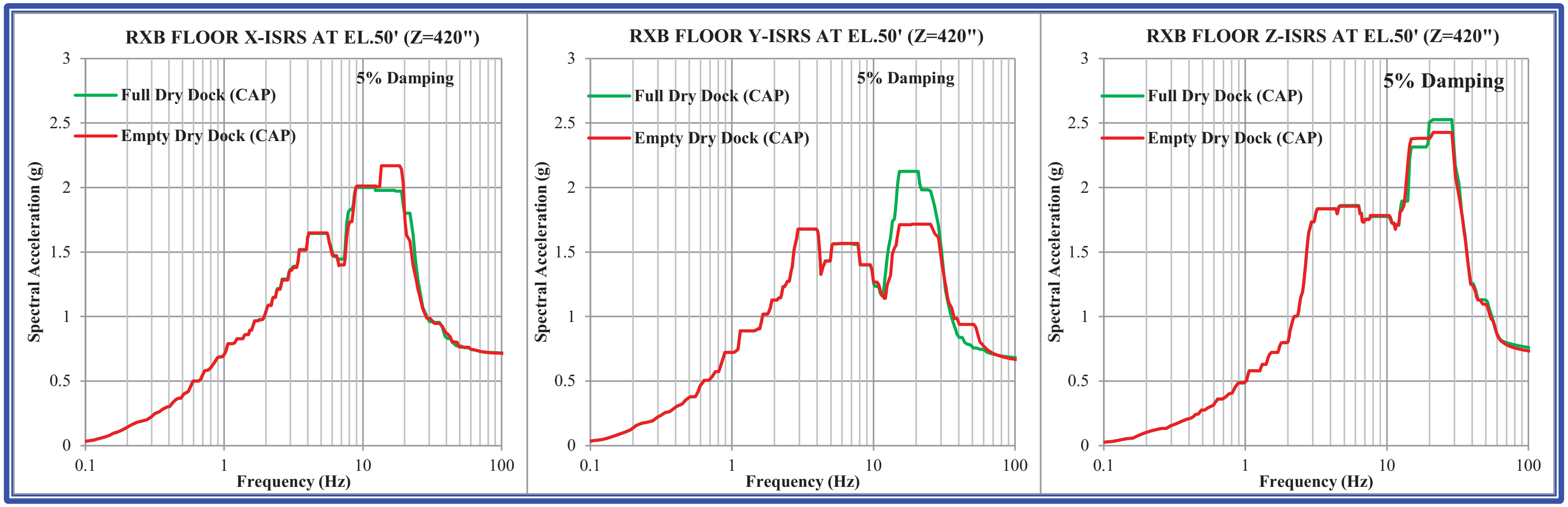
RAI 03.07.02-16S1

Table 3.7.2-60: Comparison of Maximum Empty Dry Dock Condition Forces in NuScale Power Module Skirt Supports with Final Safety Analysis Report Results (Capitola Input and Nominal NuScale Power Module Stiffness)

Dry Dock Condition	CNV Skirt E-W Reaction (kips)	CNV Skirt N-S Reaction (kips)	CNV Skirt Vertical Reaction (kips)
Full	524	455	1,625
Empty	539	452	1,645

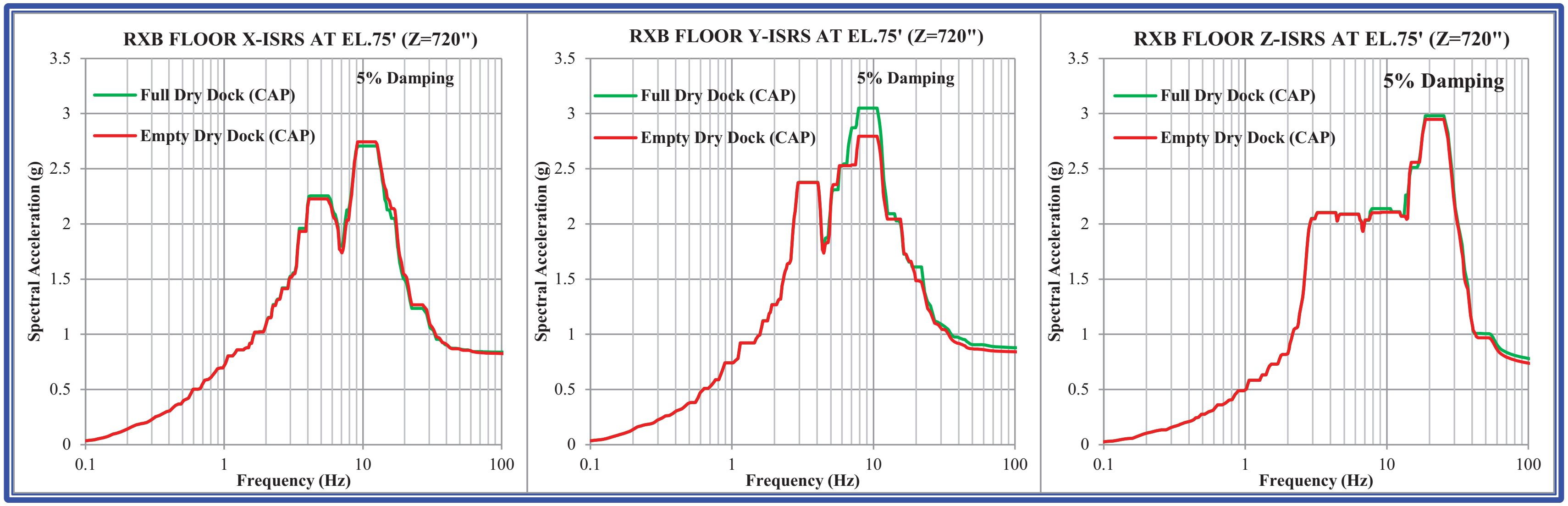
RAI 03.07.02-16S1

Figure 3.7.2-172: Floor In-Structure Response Spectra at El. 50 ft, (Z=420 in.) Comparing Full and Empty Dry Dock Conditions



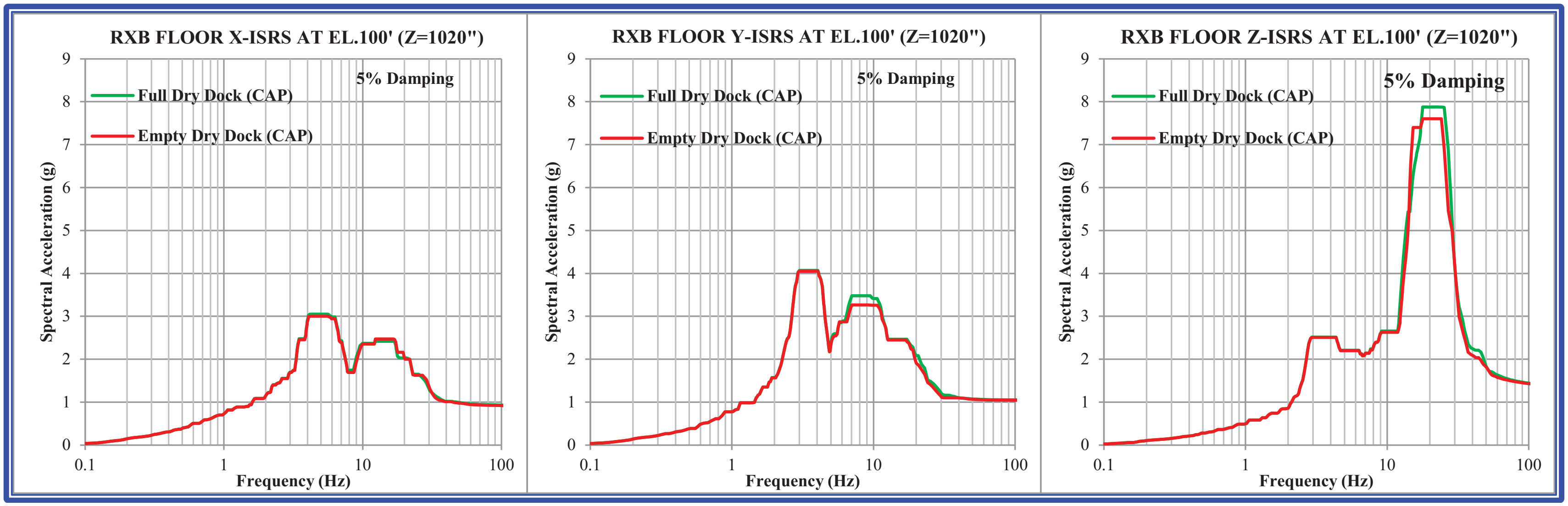
RAI 03.07.02-16S1

Figure 3.7.2-173: Floor In-Structure Response Spectra at El. 75 ft, (Z=720 in.) Comparing Full and Empty Dry Dock Conditions



RAI 03.07.02-16S1

Figure 3.7.2-174: Floor In-Structure Response Spectra at El. 100 ft, (Z=1020 in.) Comparing Full and Empty Dry Dock Conditions



RAI 03.07.02-16S1

Figure 3.7.2-175: In-Structure Response Spectra at Reactor Building Crane Wheel Node 29545 Comparing Full and Empty Dry Dock Conditions