NuScaleDCRaisPEm Resource

From:	Chowdhury, Prosanta
Sent:	Monday, April 9, 2018 3:18 PM
To:	Request for Additional Information
Cc:	Lee, Samuel; Cranston, Gregory; Tabatabai, Omid; Lupold, Timothy; Wong, Yuken; NuScaleDCRaisPEm Resource
Subject:	Request for Additional Information No. 410 eRAI No. 9310 (03.09.02)
Attachments:	Request for Additional Information No. 410 (eRAI No. 9310).pdf

Attached please find NRC staff's request for additional information (RAI) concerning review of the NuScale Design Certification Application.

Please submit your technically correct and complete response within 60 days of the date of this RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

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301-415-1647

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Request for Additional Information No. 410 (eRAI No. 9310)

Issue Date: 04/09/2018
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components
Application Section: 3.9.2

QUESTIONS

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.

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.

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.

03.09.02-62

In the response to Subquestion 2 and 3 of RAI 8911, Question 03.09.02-27, the applicant stated that Belleville washers are placed below the lower core plate to tune the core's vertical natural frequency away from high vertical NPM acceleration frequencies that has peak near 17 Hz. A target core vertical frequency of 6 Hz was selected. To achieve the target frequency, the combined spring constant of the 10 Belleville washers acting in series at each core support block was calculated to be 106,800 lbf/in. However, EC-A010-2322-R2, "Reactor Module Seismic Model" Section 2.1.5 stated that four Belleville washers are assumed to exist at bottom of core support with spring constant of 1.068E5 lbf/in for each of the four washers. The staff is not clear how the combined spring constant of 106,800 lbf/in is obtained for the 10 Belleville washers acting in series. Provide the following information:

1. Spring constant for each of the 10 Belleville washers acting in series and the formula used in calculation of the combined spring constant of 106,800 lbf/in.
2. Clarify that the Design in-structure response spectra (ISRS) of the top and lower core plates (Fig. B16 to Fig. 21 in TR-0916-51502) are calculated from the NPM seismic model with 10 Belleville washers in series, not 4 Belleville washers in series, below the lower core plate.

Include the requested information in the NPM Seismic Report.

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.

03.09.02-64

In the response to Subquestion 2 of RAI 8911, Question 03.09.02-29, the applicant stated that the upper riser bellows are between the upper riser shell and the upper riser cone section. The

bellows allow for vertical thermal growth while limiting relative horizontal deflections between the upper riser and the lower riser. The applicant further stated that while the geometry of the bellows has not been explicitly modeled, its effect has been captured by coupling the upper riser and lower riser in the horizontal directions while the vertical direction is not coupled. The staff is not clear about the properties and modeling of the bellows. Provide the following information:

1. Describe detailed properties of the bellows (thickness, connections to the upper riser including connections between the bellow and the riser sliding surfaces, sketches, etc.).
2. Does the bellow behave like a spring in axial direction? Explain why the bellows are not modelled as springs in the NPM seismic model and provide justification that the NPM seismic response without considering the spring constant of the bellows in the upper/lower riser conical joint is conservative.
3. Discuss whether sliding in the upper/lower riser conical joint will introduce nonlinearity under seismic loading. If yes, provide justification that not modeling sliding behavior in the upper/lower riser conical joint in the NPM seismic model is conservative.
4. Table C-1 of TR-0916-51502 states that the upper riser is not restrained in the vertical direction other than by gravity and compression of the bellows which keeps the interface between the upper riser and lower riser closed. Figure B-21 of TR-0916-51502 indicates that the vertical spectral frequency at high frequency end is about 1.6 g. Address the potential that the upper riser may uplift from the lower riser at the upper/lower riser conical joint under 1.6 g vertical spectral acceleration.
5. Provide stress evaluation of the upper riser bellows in the response to RAI 8911, Question 03.09.02-18 which is scheduled for July 2018.

Include the requested information in the NPM Seismic Report.

03.09.02-65

In Subquestion 1 of RAI 8911, Question 03.09.02-32, the applicant was asked why there is no simplified steam generator (SG) model within the 3D NuScale power module (NPM) seismic model. In the RAI response, the applicant stated that radial coupling between SG tube support bar assemblies, upper riser, and reactor pressure vessel (RPV) shell is provided for load transfer and the stiffness of the SG assembly in the other directions is inherently flexible and therefore, stiffness of SG is not considered in the 3D NPM seismic model. The staff noticed in Table 4-1 of EC-A014-3306 “Steam Generator Structural Model” Rev. 2 that the major modes of the SG/riser/vessel model are { }Hz (riser/SG beam mode), { }Hz (SG mode), { }Hz (SG twisting mode), { }Hz (vessel beam mode), { }Hz (riser/SG vertical mode), and { }Hz (complex mode of SG supports). EC-A014-3306 also indicates that a single helical SG tube has frequencies of the local beam mode ranging from { }Hz to { }Hz depending on location of the tube. Both cases indicate that the SG assembly is not flexible. The applicant is requested to perform a confirmative analysis to demonstrate that ignoring the SG stiffness has no significant impact on the results presented in TR-0916-51502. Alternatively, consider incorporating a simplified SG model in the 3D NPM model, similar to the simplified fuel assembly beam model incorporated in the 3D NPM model.

Include the requested information in the NPM Seismic Report.

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.

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.

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.

03.09.02-69

In the response to RAI 8911, Question 03.09.02-47, the applicant stated that although the NPM piping is being analyzed, measured and inspected under the comprehensive vibration assessment program (CVAP), NuScale also agrees that the provisions in the American Society of Mechanical Engineers (ASME) Operations and Maintenance Standard, 2012 Edition (OM-2012), Part 3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems" can be followed for piping that needs testing per the NuScale CVAP. The NRC staff finds that the selection of piping systems for vibration testing solely based on the results of the CVAP not acceptable. The applicant has the option to analyze and test the NuScale Power Module (NPM) piping under the CVAP per RG 1.20; however this does not alleviate screening and testing the NPM piping systems per the provisions of Part 3 of ASME OM-2012. The criteria for selecting the piping systems for vibration testing should be based on the provisions in Part 3 of ASME OM-2012, and not based on the results of CVAP alone. If the CVAP results identify additional piping systems than those prescribed in Part 3 of ASME OM-2012 for vibration testing, NuScale can choose to voluntarily perform vibration testing on these additional piping systems.

Revise the description in DCD Tier 2, Section 3.9.2.1. Additionally, include in DCD Tier 2, Section 14.2 a piping vibration test during initial startup. Alternatively, modify Test #97 – Thermal Expansion Test to include piping vibration testing.

03.09.02-70

In the response to Subquestion 2 of RAI 8911, Question 03.09.02-24, the applicant stated that the core barrel and reflector are modeled within the lower reactor vessel internal (RVI) submodel. The core barrel and reflector are modeled in ANSYS as shell elements and solid elements, respectively. However, the fluid gap between the core barrel and reflector is not considered in the modelling. The NRC staff noted that the narrow fluid gap { } between the core barrel and reflector can affect the natural frequency of the core barrel and reflector significantly. Without considering the fluid gap, the core barrel and reflector have fundamental frequency of { }Hz and { }Hz, respectively (data from EC-A023-3535, Rev. 0, "RVI Turbulent Buffeting Degradation Evaluation"). With the fluid gap considered in the modelling, frequency of the first five modes of the core barrel-fluid gap-reflector coupled system are { }Hz (data from ER-A010-2157, Rev.0, "Methodology Development for Hydrodynamic Effect Evaluation for Reflector and Core Barrel"). Justify that omission of the fluid gap between the core barrel and reflector in the lower RVI submodel provides conservative results of the NPM seismic response.

Include the requested information in the NPM Seismic Report.

03.09.02-71

10 CFR 50, Appendix A, GDC 2 requires systems, structures, and components important to safety be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena including earthquake. During the audit for the NuScale power module seismic analysis, the NRC staff noticed that the blowdown loads of main steam pipe break (MSPB) and feedwater pipe break (FWPB) are not included in the load combination of reactor primary vessel (RPV) primary stress calculation documented in EC-A011-2278-00 Rev. 0, “Reactor Pressure Vessel Primary Stress Calculation.” The applicant stated during the audit that blowdown analysis of MSPB and FWPB will be performed in the second quarter of 2018 and a confirmatory analysis will be conducted to demonstrate that the blowdown loads of MSPB and FWPB are bounded by the design basis pipe break (DBPB) load. If not bounded, reevaluation of RPV primary stress will be performed. The staff requests the applicant to provide the comparison of the loads between DBPB, MBPB, and FWPB at various locations of the RPV and reactor vessel internals when the blowdown loads of MSPB and FWPB are available. Also provide the updated RPV primary stress calculation if reevaluation is necessary.

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.