NuScaleDCRaisPEm Resource

From: Chowdhury, Prosanta
Sent: Tuesday, March 13, 2018 10:25 AM
To: Request for Additional Information
Cc: Lee, Samuel; Cranston, Gregory; Tabatabai, Omid; Lupold, Timothy; Wong, Yuken; NuScaleDCRaisPEm Resource
Subject: Request for Additional Information No. 386 eRAI No. 9316 (3.9.2)
Attachments: Request for Additional Information No. 386 (eRAI No. 9316).pdf

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

The NRC Staff recognizes that NuScale has preliminarily identified that the response to one or more questions in this RAI is likely to require greater than 60 days. NuScale is expected to provide a schedule for the RAI response by email within 14 days.

If you have any questions, please contact me.

Thank you.

Prosanta Chowdhury, Project Manager
Licensing Branch 1 (NuScale)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
301-415-1647

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Created By: Prosanta.Chowdhury@nrc.gov

Recipients:
"Lee, Samuel" <Samuel.Lee@nrc.gov>
Tracking Status: None
"Cranston, Gregory" <Gregory.Cranston@nrc.gov>
Tracking Status: None
"Tabatabai, Omid" <Omid.Tabatabai-Yazdi@nrc.gov>
Tracking Status: None
"Lupold, Timothy" <Timothy.Lupold@nrc.gov>
Tracking Status: None
"Wong, Yuken" <Yuken.Wong@nrc.gov>
Tracking Status: None
"NuScaleDCRaisPEm Resource" <NuScaleDCRaisPEm.Resource@nrc.gov>
Tracking Status: None
"Request for Additional Information" <RAI@nuscalepower.com>
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Request for Additional Information No. 386 (eRAI No. 9316)
Issue Date: 03/13/2018
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

03.09.02-50

In the response to RAI 8884, Question 03.09.02-2 the detailed vortex shedding evaluations of screened components were discussed. Please add the information to the comprehensive vibration assessment (CVAP) report TR-0716-50439. Without the information, the staff cannot reach a reasonable assurance finding on the structural integrity of the reactor internals components to withstand the adverse effects of vibration.

03.09.02-51

In the response to RAI 8884, Question 03.09.02-3, the staff finds that removal of the previously planned control rod assembly guide tube (CRAGT) FIV testing needs quantitative detailed justification. The very low cited margin of the CRAGT to flow-induced vibration (FIV) induced wear is based on an unverified assumed forcing function (NuScale referred to their assumption discussion in EC-A023-3535, Rev. 0, “RVI Turbulent Buffeting Evaluation”). Explain quantitatively how fluid flow estimates within and around the CRAGT (particularly through the CRAGT side holes), along with flow-induced forcing functions and vibration response are confirmed to be conservative.

Add the information to the comprehensive vibration assessment (CVAP) report TR-0716-50439.

03.09.02-52

In the response to RAI 8884, Question 03.09.02-4, the staff finds that using pre-verified software like ANSYS does not ensure valid structural models have been developed. Meshing procedures and densities, boundary condition assumptions, and fluid loading effects all should be appropriate and conservative for a given model to be reasonable and bounding. Provide a summary of the SIET TF1 and TF2 testing results for staff review and address any impacts of the results on SG FIV evaluations, wear, and fatigue life. Alternatively, NuScale may propose other options to resolve the staff’s concerns. Without the validation of finite element modelling procedures and assessment of the overall modelling approach, the staff cannot reach a reasonable assurance finding on the structural integrity of the reactor internals components to withstand the adverse effects of vibration because finite element modelling procedures and the overall FIV modelling approach have not been demonstrated to be conservative, and as such could lead to failure of the reactor vessel internals. Add a summary of the information and any key conclusions regarding SG FIV to the comprehensive vibration assessment (CVAP) report TR-0716-50439.

03.09.02-53

In the response to RAI 8884, Question 03.09.02-5, the table of velocities added to the markup CVAP report is incomplete. The table is limited, and does not include velocities for all evaluated components and FIV mechanisms. As originally requested, add detailed velocity information to the CVAP report for all evaluated components and FIV mechanisms, including fluid-elastic instability (FEI), VS, turbulence buffeting (TB), and acoustic resonance (AR). Also include velocities computed from thermal hydraulic analysis, being sure to include those used for steam generator (SG) tube and SG support bar VS and FEI, as well as DHRS AR assessments. Include any available bias errors and uncertainties based on existing thermal hydraulic test data. Without the information, the staff cannot reach a reasonable assurance finding on the structural integrity of the reactor internals components to withstand the adverse effects of vibration because the velocity is a key parameter in the flow-induced vibration analysis and the application must contain a level of design information sufficient to enable the staff to judge the applicant's proposed design.

03.09.02-54

Provide the detailed information on acoustic resonance (AR) assessments to the CVAP report as originally requested in RAI 8884, Question 03.09.02-7, particularly on the decay heat removal system (DHRS) standby modes. Given that strong acoustic resonances excited by second order shear layer flow instabilities have been observed in the main steam lines of boiler water reactors (BWRs) it has been common practice for licensees requesting Extended Power Uprates (EPUs) or those proposing new plant designs to confirm that first and second order shear layer instabilities do not lock-on to sidebranch acoustic modes and damage valves or generate acoustic loads which could damage other components including reactor internals. Other evidence of the ability of 2nd order instabilities to excite acoustic resonances are available in Ziada and Lafon, "Flow-excited acoustic resonance excitation mechanism, design guidelines, and counter-measures," Applied Mechanics Reviews, Vol. 66, Jan 2014. Expand the NuScale AR assessments, particularly in the DHRS, to include second order instabilities. Also, given the small margin against AR at the primary shear layer modes, explain in detail how AR will be assessed and, if necessary, mitigated during initial startup or other testing. Without the information, the staff cannot reach a reasonable assurance finding on the structural integrity of the reactor internals components to withstand the adverse effects of vibration.

Add the information to the comprehensive vibration assessment (CVAP) report TR-0716-50439.

03.09.02-55

In RAI 8884, Question 03.09.02-8, the staff requested NuScale to provide the separate effects test results of the steam generator inlet flow restrictor (SGIFR). The response indicates that the SGIFR testing is to be re-performed. Provide a description of the final SGIFR design and a summary of the test plan. Without the information, the staff cannot reach a reasonable assurance finding on the structural integrity of the reactor internals components to withstand the adverse effects of vibration because the SGIFR is susceptible to leakage flow instability and there is no analysis performed.

03.09.02-56

In the response to RAI 8884, Question 03.09.02-13, NuScale provided quantitative criteria for evaluating structures for fatigue due to turbulence buffeting and markup changes to the CVAP report. The staff requests that the applicant add details on the turbulent buffeting evaluations of components with lowest resonance frequencies less than 200 Hz to the CVAP report. Without the information, the staff cannot reach a reasonable assurance finding on the structural integrity of the reactor internals components to withstand the adverse effects of vibration.

03.09.02-57

In the response to RAI 8740, Question 14.02-1, NuScale revised the DCD Tier 2, Chapter 14.2 descriptions for Tests #44, #45, and #72 from startup testing to factory tests or separate effects tests. DCD Tier 2, Section 14.2 is for preoperational and initial startup testing. As indicated in DCD Tier 2, Section 14.2, NuScale is not planning to conduct preoperational testing for reactor internals. Startup testing is defined as testing performed in the assembled plant after fuel load. However, NuScale listed the following factory tests or special effects tests in Chapter 14.2:

- Test #44: Control Rod Drive System Flow-Induced Vibration Test performed during factory testing
- Test #45: Reactor Vessel Internals Flow-Induced Vibration Test (only in-core instrument guide tubes) performed at the factory
- Test #72: Steam Generator Flow-Induced Vibration Test [These are actually the separate effects tests on steam generator tube columns (but with only a limited number of tube columns and possibly no flow inside the SG tubes), and steam generator inlet flow restrictors]

These are tests that will be performed in a laboratory or factory, and not in the assembled plant as a hot functional test or initial startup test. Therefore, relocate these tests from DCD Tier 2, Section 14.2, or provide justification for including these tests in Chapter 14.2.

03.09.02-58

10 CFR 50, Appendix A, GDC 4 requires structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. DCD, Tier 2, Table 1.9-2, comments for RG 1.20 state the following:

> The aspects of this RG that mandate start-up tests or inspections for non-prototype reactors are applicable as these tests must be accommodated in the start-up of each installed NuScale Power Module. The remainder is applicable to the COL applicant, and only specific COL applicants which are for prototype reactors.

The comments are not clear. Provide clarification on these statements and the exceptions to RG 1.20. Revise the DCD if necessary.