NuScaleDCRaisPEm Resource

From: Cranston, Gregory
Sent: Wednesday, June 28, 2017 11:53 AM
To: RAI@nuscalepower.com
Cc: NuScaleDCRaisPEm Resource; Lee, Samuel; Chowdhury, Prosanta; Mitchell, Matthew; Widrevitz, Dan; Bavol, Bruce
Subject: Request for Additional Information No. 75, RAI 8904
Attachments: Request for Additional Information No. 75 (eRAI No. 8904).pdf

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

Please submit your 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.

Gregory Cranston, Senior Project Manager
Licensing Branch 1 (NuScale)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
301-415-0546

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Recipients:
"NuScaleDCRaisPEm Resource" <NuScaleDCRaisPEm.Resource@nrc.gov>
Tracking Status: None
"Lee, Samuel" <Samuel.Lee@nrc.gov>
Tracking Status: None
"Chowdhury, Prosanta" <Prosanta.Chowdhury@nrc.gov>
Tracking Status: None
"Mitchell, Matthew" <Matthew.Mitchell@nrc.gov>
Tracking Status: None
"Widrevitz, Dan" <Dan.Widrevitz@nrc.gov>
Tracking Status: None
"Bavol, Bruce" <Bruce.Bavol@nrc.gov>
Tracking Status: None
"RAI@nuscalepower.com" <RAI@nuscalepower.com>
Tracking Status: None

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Request for Additional Information No. 75 (eRAI No. 8904)

Issue Date: 06/28/2017
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 04.05.02 - Reactor Internal and Core Support Structure Materials
Application Section: 4.5.2

QUESTIONS

04.05.02-1

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria 1 and 10 CFR Part 50.55a require that structures, systems, and components (SSCs) important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed.

NuScale DCD Section 4.5.2.5 states that washers in the RVI upper riser assembly are made from Alloy 718. There is no corresponding text regarding Alloy 718 nuts. The section refers to DCD Section 3.13.1 for discussion concerning the annealing and precipitation hardening treatment for Alloy 718 materials. DCD Section 4.5.2.5 does not address the potential for stress corrosion cracking (SCC) for Alloy 718 materials exposed to reactor coolant. DCD Section 3.13 does not address the potential for SCC in components exposed to high temperatures and reactor coolant.

The staff requests that the applicant address the potential for SCC of Alloy 718 washers and nuts in the RVI upper riser assembly and revise the DCD accordingly.

04.05.02-2

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria 1 and 10 CFR Part 50.55a require that structures, systems, and components (SSCs) important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed.

NuScale DCD Section 4.5.2.1 states that,

> Neutron irradiation-induced degradations such as irradiation-assisted stress corrosion cracking, void-swelling, stress-relaxation, and irradiation embrittlement have been evaluated using material aging degradation mechanism screening criteria of the Electric Power Research Institute (EPRI) materials reliability program (Reference 4.5-3).

The applicant then states that,

> The components meeting the screening criteria are the upper CRDS [control rod drive system] supports, ICI [in-core instrumentation] guide tube supports, CRDS alignment cones, CRA [control rod assembly] lower flange, CRA cards, fuel pins, shared fuel pins, fuel pin caps, upper support blocks, lower core plate alignment pins, lower core support lock insert, core support lock plate assembly, and the lock plate. In addition, components identified as susceptible to irradiation-induced stress relaxation are also included for potential wear due to loosening.

No discussion is presented regarding what, if anything, is done regarding the components that meet the criteria, or which criteria were met for which components. The staff notes that although MRP-175, "PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values," was used to provide the criteria for the NuScale evaluation, the results of applying MRP-175 based criteria do not stand alone from how the evaluation was conducted for NuScale specifically. The staff has been provided with no basis with which to verify the evaluation and no consequent actions for components which were deemed susceptible.

The staff requests that, (a) the applicant provide the evaluations that were performed, and (b) a discussion of the compensatory actions applied (or to be applied) to the components that screened in. If no compensatory actions are required, the staff request that this be justified in the response to (b). The staff further request that the response to (b) be added to the DCD.

04.05.02-3

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria 1 and 10 CFR Part 50.55a require that structures, systems, and components (SSCs) important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed.

NuScale DCD Section 4.5.2 does not address the potential for crevice corrosion. Crevice corrosion has proven a significant factor for internals aging management in the operating fleet.

The staff requests that NuScale provide any assessment pertaining to the potential for crevice corrosion in core support and reactor internal components. The staff notes that significant discussion of this topic was included in DCD Subsection 5.4.1.2 for steam generator components.

The staff further requests that a discussion of crevice corrosion be included in DCD Subsection 4.5.2 unless the applicant can establish that the potential for crevice corrosion for the subject components is too low to merit inclusion.