From:	Getachew Tesfaye
Sent:	Friday, March 15, 2024 12:24 PM
To:	Request for Additional Information
Cc:	Prosanta Chowdhury; Mahmoud -MJ- Jardaneh; Griffith, Thomas; Osborn, Jim; Fairbanks, Elisa; NuScale-SDA-720RAIsPEm Resource
Subject:	NuScale SDAA Section 3.9.4 - Request for Additional Information No. 015 (RAI-10133-R1)
Attachments:	SECTION 3.13 - RAI-10133-R1-FINAL.pdf

Attached please find NRC staff's request for additional information (RAI) concerning the review of NuScale Standard Design Approval Application for its US460 standard plant design (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23306A033).

Please submit your technically correct and complete response by the agreed upon date to the NRC Document Control Desk.

If you have any questions, please do not hesitate to contact me.

Thank you,

Getachew Tesfaye (He/Him)
Senior Project Manager
NRC/NRR/DNRL/NRLB
301-415-8013

Hearing Identifier: NuScale_SDA720_RAI_Public
Email Number: 28

Mail Envelope Properties (BY5PR09MB5682FC32E103D6368760C1418C282)

Subject: NuScale SDAA Section 3.9.4 - Request for Additional Information No. 015 (RAI-10133-R1)
Sent Date: 3/15/2024 12:23:33 PM
Received Date: 3/15/2024 12:23:39 PM
From: Getachew Tesfaye

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Post Office: BY5PR09MB5682.namprd09.prod.outlook.com

Files	**Size**	**Date & Time**
MESSAGE	584	3/15/2024 12:23:39 PM
SECTION 3.13 - RAI-10133-R1-FINAL.pdf	146099	

Options
Priority: Normal
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:

REQUEST FOR ADDITIONAL INFORMATION No. 015 (RAI-10133-R1)
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
NUSCALE STANDARD DESIGN APPROVAL APPLICATION
DOCKET NO. 05200050
CHAPTER 3, "DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS AND EQUIPMENT"
SECTION 3.13, "THREADED FASTENERS (AMERICAN SOCIETY OF MECHANICAL ENGINEERS CODE CLASS 1, 2, AND 3)"
ISSUE DATE: 03/15/2024

Background

By letter dated October 31, 2023, NuScale Power, LLC (NuScale or the applicant) submitted Part 2, Final Safety Analysis Report (FSAR), Chapter 3, "Design of Structures, Systems, Components and Equipment," Revision 1 (Agencywide Documents Access and Management System Accession No. ML23304A321), of the NuScale Standard Design Approval Application (SDAA) for its US460 standard plant design. The applicant submitted the US460 standard plant SDAA in accordance with the requirements of Title 10 *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Subpart E, "Standard Design Approvals." The NRC staff has reviewed the information in Chapter 3 of the SDAA and determined that additional information is required to complete its review.

Question 3.13-1

Regulatory Basis

- 10 CFR 50.55a Codes and standards: ECCS Reactor Vent Valve (RVV) and Reactor Recirculation Valve (RRV) connections to Reactor Pressure Vessel (RPV) are ASME Code Section III Class 1 connections, and therefore must meet the ASME Section III Subsection NB requirements per 10 CFR 50.55a.
- 10 CFR Part 50 Appendix-A, GDC 4, "Environmental and dynamic effects design bases," states that structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.
- 10 CFR 50.46 Acceptance criteria for emergency core cooling systems (ECCS) for light-water nuclear power reactors describe, in part, that emergency core cooling systems cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.

Issue

Standard Review Plan (SRP) Section 3.13, "Threaded Fasteners - ASME Code Class 1, 2, and 3," provides guidance for reviewing and evaluating the adequacy of an applicant's criteria in regard to selection of materials, design, inspection, and testing of its threaded fasteners (i.e.,

threaded bolts, studs, etc.) prior to initial service and during service in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Class 1, 2 or 3 systems.

The staff noted that a change in the design of connections with Reactor Pressure Vessel from the DCA to the SDA. The DCA design utilized primarily welded connections with a few bolted connections for a limited number of connections, while the SDAA design utilizes primarily bolted connections with threaded inserts. This change impacts safety-related ASME BPV Code Section III Class 1 components that constitute the reactor coolant pressure boundary. Due to the safety significant nature of these connections and the significant change in the SDAA design from that reviewed in the DCA, the staff needs additional information to support its safety review of the structural integrity of these connections.

The RVV and RRV connections to the RPV are the most safety significant valve connections in the ECCS. Gross failure at these locations would result in a significant challenge to fuel integrity and would result in unanalyzed dynamic loads that may challenge the containment boundary. If breaks at these connections are not assumed, then appropriate justification must be provided. Using risk-informed considerations, the decrease in defense-in-depth due to not postulating losses of coolant at these locations can be compensated, in part, by a high confidence in the structural integrity of the bolted connections. According to GDC 4, dynamic effects associated with postulated pipe ruptures can be excluded from the design basis when analyses that demonstrate that the probability of fluid system piping rupture is extremely low under design basis conditions are reviewed and approved by the Commission. This can be satisfied, in part, by confirming that stress and cumulative usage factors (CUF) margins meet at least the BTP3-4 criteria using acceptable analysis methods, having appropriate augmented measures in materials, fabrication, and preservice and inservice inspections, and continuing to meet the guidelines and recommendations in EPRI NP-5769 [Degradation and failure of Bolting in Nuclear Power Plants].

During an audit meeting, NuScale stated that they would complete a Finite Element Analyses (FEA) to determine the stress and CUF values for bolts, threaded inserts, and other components in these connections, but has indicated that this analysis would not be completed until the ITAAC phase. The staff does not consider regulatory commitments to complete the design evaluation and meet the ASME BPV Code acceptance criteria through an ITAAC as an appropriate resolution for these highly safety significant ECCS RVV and RRV connections due to the far-reaching implications from 10 CFR Appendix A, GDC 4, and 10 CFR 50.46 perspectives.

In order to reach a safety finding for the SDAA review, the NRC is requesting a summary of results for the current design configuration' Due to as-built configurations, the NRC realizes that minor changes may occur in the final design of ECCS valve bolted connections. Therefore, the final as-built FEA analysis or a reconciliation for the ECCS Valve bolted connections analyses can be submitted during the ITAAC phase.

Information Requested

Provide a summary of the method and assumptions used in the FEA (or other analysis methodology if FEA is not used). Additionally, provide a summary of preliminary analysis results of representative bolted connection from each group (RSV, RRV, RVV, I&C, feed plenum access port, PZR Heater cover, and MS Plenum). Include ASME BPV Code Section III Class 1 stress

intensities and CUFs for all the key components of the bolted connections (bolting, threaded inserts, flanges, and others), based on the current design of the bolted connections. This FEA evaluation (or other analysis methodology) should be made available for staff audit. The discussion should address the inputs, boundary conditions, and margins in stresses and CUF values for threaded inserts, bolts, flanges, and other key components. Revise the FSAR as appropriate with the summary of the analysis results.