NuScaleDCRaisPEm Resource

From:	Cranston, Gregory
Sent:	Friday, May 19, 2017 2:58 PM
To:	'RAI@nuscalepower.com'
Cc:	NuScaleDCRaisPEm Resource; Lee, Samuel; Chowdhury, Prosanta; Hayes, Michelle; Nakanishi, Tony; Bavol, Bruce; Franovich, Rani
Subject:	Request for Additional Information No. 25 (RAI 8813)
Attachments:	Request for Additional Information No. 25 (eRAI No. 8813).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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From: Cranston, Gregory

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Request for Additional Information No. 25 (eRAI No. 8813)

Issue Date: 05/19/2017
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19

QUESTIONS

19-1

10 CFR 52.47(a)(27) states that a DC application must contain a Final Safety Analysis Report (FSAR) that includes a description of the design-specific probabilistic risk assessment (PRA) and its results. In accordance with the Statement of Consideration (72 FR 49387) for the revised 10 CFR Part 52, the staff reviews the information contained in the applicant's FSAR Chapter 19, and issues requests for additional information (RAI) and conducts audits of the complete PRA (e.g., models, analyses, data, and codes) to obtain clarifying information as needed. The staff uses guidance contained in Standard Review Plan (SRP) Chapter 19.0 Revision 3, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors." In accordance with SRP Chapter 19.0 Revision 3, the staff determines whether:

"The PRA reasonably reflects the as-designed, as-built, and as-operated plant, and the PRA maintenance program will ensure that the PRA will continue to reflect the as-designed, as-built, and as-operated plant, consistent with its identified uses and applications."

The staff has reviewed the information in the FSAR and examined additional clarifying information from the audit of the complete PRA and determined that it needs additional information to confirm that the PRA reasonably reflects the as-designed plant. Specifically, the staff is unclear if the emergency core cooling system (ECCS) model includes all important failure modes for systems, structures and components (SSCs) identified as risk significant by the applicant. Based on its review the staff believes that the potential failure of the inadvertent actuation block (IAB) feature is not explicitly modeled in the ECCS-T01 top event (reactor vent valves and reactor recirculation valves open). If the IAB fails closed, the ECCS main valve would fail to open. The staff has confirmed that the IAB failing to close is modeled for spurious opening of the ECCS main valve.

a) The staff requests the applicant to explain how the IAB failing closed when ECCS actuation is called upon is accounted for in the model.

b) Similarly, the staff requests the applicant to explain how it reached the conclusion that ECCS trip line plugging as an ECCS failure mode is not credible based on its design and the cleanliness of the reactor coolant.