NuScaleDCRaisPEm Resource

From: Cranston, Gregory
Sent: Thursday, August 10, 2017 3:32 PM
To: RAI@nuscalepower.com
Cc: NuScaleDCRaisPEm Resource; Lee, Samuel; Chowdhury, Prosanta; Hayes, Michelle; Nakanishi, Tony; Franovich, Rani
Subject: Request for Additional Information No. 160, RAI 8982 (19)
Attachments: Request for Additional Information No. 160 (eRAI No. 8982).pdf

Attached please find NRC staff's request for additional information 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.

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. 160 (eRAI No. 8982)

Issue Date: 08/10/2017
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

19-25

10 CFR 52.47(a)(27) states that a design certification application must contain an final safety analysis report (FSAR) that includes description of the design-specific probabilistic risk assessment (PRA) and its results. In accordance with the Statement of Consideration (72 Federal Register 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 applicant has performed sensitivity studies sufficient to gain insights about the impact of uncertainties (and the potential lack of detailed models) on the estimated risk. The objectives of the sensitivity studies should include (1) determining the sensitivity of the estimated risk to potential biases in numerical values, such as initiating event frequencies, failure probabilities, and equipment unavailabilities, (2) determining the impact of the potential lack of modeling details on the estimated risk, and (3) determining the sensitivity of the estimated risk to previously raised issues (e.g., motor-operated valve reliability). As noted in Element 1.1 of Table A-1 in Appendix A to RG 1.200, *special emphasis should be placed on PRA modeling of novel and passive features in the design*, as well as addressing issues related to those features, such as DI&C, explosive (squib) valves, and the issue of T-H [thermal-hydraulic] uncertainties." (Emphasis added)

In addition, SRP Chapter 19.0, Revision 3, directs the staff to determine whether "The applicant has used the PRA results and insights, including those from uncertainty analyses, importance analyses, and sensitivity studies, in an integrated fashion."

The staff notes that the FSAR includes sensitivity analysis results for select PRA modeling assumptions. However, to gain sufficient insights about the impact of uncertainties on the estimated risk, the staff requests the applicant to augment its sensitivity analyses to include additional evaluation of assumptions associated with reliability of design features that are unique and risk-significant. The staff requests this information in light of limited availability or unavailability of design-specific operating experience of these design features at this stage of the design. The staff will use the information to assess the uncertainties in NuScale risk estimates introduced by applying data from generic large light water reactor experience to the NuScale design and operating environment.

a) Please evaluate the sensitivity of the core damage frequency and large release frequency to the failure probabilities associated with design features for the following safety functions:

1. emergency core cooling
2. containment isolation
3. control rod insertion
4. module protection system functions

b) Please discuss the extent to which the sensitivity analysis results have been used in an integrated fashion and factored into the determination of the risk-significant SSCs and human actions.