NuScaleDCRaisPEm Resource

From:	Cranston, Gregory
Sent:	Monday, July 10, 2017 11:51 AM
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Cc:	NuScaleDCRaisPEm Resource; Lee, Samuel; Chowdhury, Prosanta; Hayes, Michelle; Schaperow, Jason; Franovich, Rani
Subject:	Request for Additional Information No. 92, RAI 8903
Attachments:	Request for Additional Information No. 92 (eRAI No. 8903).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

Hearing Identifier: NuScale_SMR_DC_RAI_Public
Email Number: 110

Mail Envelope Properties (a8e4189d866a4c4086fd21edb0e81b5f)

Subject: Request for Additional Information No. 92, RAI 8903
Sent Date: 7/10/2017 11:51:07 AM
Received Date: 7/10/2017 11:51:10 AM
From: Cranston, Gregory

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Post Office: HQPWMSMRS08.nrc.gov

Files	**Size**	**Date & Time**
MESSAGE	522	7/10/2017 11:51:10 AM
Request for Additional Information No. 92 (eRAI No. 8903).pdf	91688	

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

Request for Additional Information No. 92 (eRAI No. 8903)

Issue Date: 07/10/2017
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

19-16

Regulatory Basis

Probabilistic Risk Assessment (PRA) and Severe Accident (SA) Analysis
10 CFR 52.47(a)(27) states that a design certification application (DCA) must contain an Final Safety Analysis Report (FSAR) that includes a description of the design-specific Probabilistic Risk Assessment (PRA) and its results. 10 CFR 52.47(a)(23) states that a DCAfor light-water reactor (LWR) designs must contain an FSAR that includes a description and analysis of design features for the prevention and mitigation of severe accidents.

Design Basis Loss of Coolant Accident (DB-LOCA) Offsite Radiological Consequence Assessment
10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 10 CFR 52.47(a)(2)(iv)(A) and 10 CFR 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

Request for additional information
ER-P060-5275-R0, “Severe Accident Evaluation Supporting Design Basis Source Term,” documents the applicant’s MELCOR simulations for scenarios with spurious reactor recirculation valve (RRV) actuation. The MELCOR simulations predict the following sequence of events:

1. The water in the reactor flows into the containment as a result of the initial blowdown and subsequent boil off.
2. The core becomes uncovered, and core damage begins.
3. The water in the containment gravity drains back into the reactor reflooding the core terminating core damage and limiting the fission product release.

ER-P020-4896-R0, “Severe Accident Selection Methodology,” states that scenarios with spurious RRV actuation and chemical and volume control system (CVCS) breaks at low elevations were not selected for its PRA and SA analysis, because they allow water in the containment to reflood the core halting core damage and negating severe accident risk to the containment.

Gravity draining of water from the containment to the reactor could be important to the PRA and SA analysis and the DB-LOCA offsite radiological consequence assessment. Therefore, the applicant is requested to either

a) Show that its modeling of water draining (by gravity) from the containment to the reactor is realistic (e.g., experimental validation) or
b) Show that its PRA and SA analysis and DB-LOCA offsite radiological consequence assessment are insensitive to its modeling of gravity draining water from the containment to the reactor.