NuScaleDCRaisPEm Resource

From: Chowdhury, Prosanta
Sent: Tuesday, May 1, 2018 2:58 PM
To: Request for Additional Information
Cc: Lee, Samuel; Cranston, Gregory; Franovich, Rani; Karas, Rebecca; Thurston, Carl; NuScaleDCRaisPEm Resource
Subject: Request for Additional Information No. 451 eRAI No. 9517 (15.06.05)
Attachments: Request for Additional Information No. 451 (eRAI No. 9517).pdf

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

Prosanta Chowdhury, Project Manager
Licensing Branch 1 (NuScale)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
301-415-1647

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Created By: Prosanta.Chowdhury@nrc.gov

Recipients:
"Lee, Samuel" <Samuel.Lee@nrc.gov>
Tracking Status: None
"Cranston, Gregory" <Gregory.Cranston@nrc.gov>
Tracking Status: None
"Franovich, Rani" <Rani.Franovich@nrc.gov>
Tracking Status: None
"Karas, Rebecca" <Rebecca.Karas@nrc.gov>
Tracking Status: None
"Thurston, Carl" <Carl.Thurston@nrc.gov>
Tracking Status: None
"NuScaleDCRaisPEm Resource" <NuScaleDCRaisPEm.Resource@nrc.gov>
Tracking Status: None
"Request for Additional Information" <RAI@nuscalepower.com>
Tracking Status: None

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Request for Additional Information No. 451 (eRAI No. 9517)

Issue Date: 05/01/2018
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 15.06.05 - Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
Application Section:

QUESTIONS

15.06.05-6

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion (GDC) 35, “Emergency Core Cooling,” requires that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. DSRS Section 15.6.5 provides guidance for complying with GDC 35. It requires that evaluation models meet the requirements of 10 CFR 50.46, which states that the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident.

In FSAR Chapter 15.6.5 and Section 9 of the “Loss-of-Coolant Accident Evaluation Model,” TR-0516-49422-P, Rev. 0, which is incorporated by reference into the FSAR, indicates that a stable natural recirculation flow pattern with the reactor recirculation valves and steam venting through the reactor vent valves (RVVs) is relied upon to remove decay heat passively via boiling in the core. The staff noted that its sensitivity calculations, using the applicant’s NRELAP5 LOCA input models, show that flows in core hot and average channels appear to be artificially exaggerated by high reverse flow in the bypass channel during the recirculation phase of the LOCA, a condition not noted in the LOCA TR. This appears to be an artifact of the 1-D core and riser modeling that causes excessive recirculation cooling via the bypass.

Please provide an analysis that quantifies NRELAP5 bypass flow rates during the ECCS recirculation phase and provide any updates to the LOCA methodology needed to resolve this issue to support the staff's GDC 35 finding.