NuScaleTRRaisPEm Resource

From: Cranston, Gregory
Sent: Saturday, June 16, 2018 8:24 AM
To: Request for Additional Information
Cc: Lee, Samuel; Karas, Rebecca; Drzewiecki, Timothy; Franovich, Rani; Chowdhury, Prosanta; NuScaleTRRaisPEm Resource
Subject: Request for Additional Information Letter No. 9536 (eRAI No. 9536) ECCS, 15.6.6, SRSB
Attachments: Request for Additional Information No. 9536 (eRAI No. 9536).pdf

Attached please find NRC staff's request for additional information (RAI) concerning review of the NuScale Topical Report.

The NRC Staff recognizes that NuScale has preliminarily identified that the response to one or more questions in this RAI is likely to require greater than 60 days. NuScale is expected to provide a schedule for the RAI response by email within 14 days.

If you have any questions, please contact me.

Thank you.

Hearing Identifier: NuScale_SMR_DC_TR_Public
Email Number: 94

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Subject: Request for Additional Information Letter No. 9536 (eRAI No. 9536) ECCS, 15.6.6, SRSB
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From: Cranston, Gregory

Created By: Gregory.Cranston@nrc.gov

Recipients:
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Tracking Status: None
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Options
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Request for Additional Information No. 9536 (eRAI No. 9536)

Issue Date: 06/16/2018
Application Title: NuScale Topical Report
Operating Company: NuScale
Docket No. PROJ0769
Review Section: 15.06.06 - Inadvertent Operation of the Emergency Core Cooling System (ECCS)
Application Section: 15.6.6

QUESTIONS

15.06.06-2

GDC 10, *Reactor design*, requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). Additionally, radiological consequences of postulated accidents must meet the requirements of 10 CFR 52.47(a)(2)(iv)(A), 10 CFR 52.47(a)(2)(iv)(B), and GDC 19, *Control room*. FSAR, Tier 2, Section 4.4.1.1 states tha the design basis for the thermal-hydraulic design of the reactor core includes providing adequate heat transfer from the fuel cladding to the reactor coolant by assuring that critical heat flux limits are met during normal operation, AOOs, and infrequent event.s. FSAR, Tier 2, Section 15.6.6.3.1 describes the use of the Hench-Levy and Griffith-Zuber critical heat flux (CHF) correlations in the evaluation of the inadvertent operation of emergency core cooling system event, which is identified as an AOO in FSAR, Tier 2, Table 15.0-1. FSAR, Tier 2, Section 15.6.6.3, states that the CHF data from the KATHY test loop was used to establish a 95/95 CHF ratio (CHFR) limit of 1.122 for the Hench-Levy CHF correlation, and that data collected at Stern Laboratories was used to establish a CHFR limit of 1.37 for the Griffith-Zuber CHF correlation.

NRC staff needs to establish a finding that the proposed CHF correlations are acceptable for performing safety analyses of the Nuclear Power Module with NuFuel-HTP2™ fuel, with a specified CHFR limit, over a specified range of applicability, and in accordance with a specified methodology. Accordingly, NRC staff requests that the applicant (1) submit a methodology, for NRC staff review, that describes the experimental data supporting the development of the CHFR limits, and that demonstrates the CHF models have sufficient validation as demonstrated through appropriate quantification of error, and (2) update the appropriate licensing documentation to consistently reflect the final CHFR limits.