NuScaleDCRaisPEm Resource

From: Chowdhury, Prosanta
Sent: Wednesday, May 2, 2018 4:05 PM
To: Request for Additional Information
Cc: Lee, Samuel; Cranston, Gregory; Franovich, Rani; Karas, Rebecca; Lu, Shanlai; NuScaleDCRaisPEm Resource
Subject: Request for Additional Information No. 460 eRAI No. 9481 (15.06.05)
Attachments: Request for Additional Information No. 460 (eRAI No. 9481).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.

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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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Request for Additional Information No. 460 (eRAI No. 9481)

Issue Date: 05/02/2018
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: 15.6.5

QUESTIONS

15.06.05-8

Title 10, Part 50, Section 50.46 "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" specifies the loss of coolant accident (LOCA) evaluation model includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure. Regulatory Guide 1.203 describes a process that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in developing and assessing evaluation models (EMs) that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. NuScale Design-Specific Review Standard Section 15.6.5, "Loss-Of-Coolant Accidents Resulting From Spectrum Of Postulated Piping Breaks Within The Reactor Coolant Pressure Boundary," directs the staff to evaluate whether the appropriate break locations, break sizes, and initial conditions were selected in a manner that conservatively predicts the consequences of the LOCA for evaluating emergency core cooling system performance.

Final Safety Analysis Report Tier 2, Section 15.6.5.3.2, "Input Parameters and Initial Conditions," states that, "[reactor coolant system (RCS)] average temperature is initialized to yield a maximum riser operation temperature of 595 ^{o}F." The 595 ^{o}F value of the riser operation temperature corresponds to a T-avg of 547.5 ^{o}F based on the chemical and volume control system (CVCS) line break LOCA NRELAP5 calculation. However, Table 15.0-6 lists all the module initial condition ranges for design basis event evaluation, with the T-avg value specified as 545 ^{o}F +/- 10 ^{o}F uncertainty. Therefore, the conservative and maximum initial condition T-avg should be 555 ^{o}F with a corresponding T-hot of 605 ^{o}F. Explain why the design basis CVCS line break case did not initialize the upper bound T-hot corresponding to the maximum T-avg required by Table 15.0-6.