NuScaleTRRaisPEm Resource

From: Cranston, Gregory
Sent: Monday, May 28, 2018 11:19 AM
To: Request for Additional Information
Cc: Lee, Samuel; Karas, Rebecca; Lu, Shanlai; Franovich, Rani; NuScaleTRRaisPEm Resource; Chowdhury, Prosanta; Schmidt, Jeffrey
Subject: Request for Additional Information Letter No. 9475 (eRAI No. 9475) Topical Report, LOCA, 15.6.5, SRSB
Attachments: Request for Additional Information No. 9475 (eRAI No. 9475).pdf

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

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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Request for Additional Information No. 9475 (eRAI No. 9475)

Issue Date: 05/28/2018
Application Title: NuScale Topical Report
Operating Company: NuScale
Docket No. PROJ0769
Review Section: 15.06.05 - Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
Application Section: NuScale LOCA Topical Report Review

QUESTIONS

15.06.05-17

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 (EM) 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 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 (DSRS) 15.6.5, "Loss-Of-Coolant Accidents Resulting From Spectrum Of Postulated Piping Breaks Within The Reactor Coolant Pressure Boundary," requires the staff to evaluate whether appropriate break locations, break sizes and initial conditions in a conservative manner that the consequences of the LOCA are fully evaluated.

In Section 5.3 of NuScale Power Topical Report TR-0516-49422, "LOCA Evaluation Model," which is incorporate by reference into the final safety analysis report (FSAR), the applicant listed all the process parameters associated with plant initial conditions. Although all the process parameters are intended to be conservatively biased, it is not clear how each individual parameter will be biased. More detailed information is needed for the following parameters:

a. Reactor coolant system (RCS) average temperature at operating conditions

It is not clear to the staff how the RCS average temperature would be maximized within the context of all the relevant analysis limits and technical specifications. Are there any technical specifications that limit the upper bound value of this parameter? For each selected RCS average temperature value, will the corresponding riser fluid temperature be consistent with the selected maximum riser fluid temperature value? Describe the method and the basis for selecting the maximum value of this parameter and a sample upper bound value based on the existing FSAR information.

b. Riser temperature

It is not clear to the staff how the riser temperature will be maximized within the context of all the relevant analysis limits, trip set points and technical specifications. For a given riser temperature value, there is a corresponding RCS average temperature value. How can the initial steady state calculation be performed to reach the selected maximum riser temperature and maximum RCS average temperature at the same time? Describe the method and the basis of selecting the maximum value of this parameter and a sample upper bound value based on the existing FSAR information.

c. Pressurizer pressure

In Table 5-6, it states that maximum pressure is applied. How will this maximum pressure value be selected? Are there any trip set points, technical specification limits or initial condition ranges which provide the basis of selecting the maximum value of this parameter? Describe the method and the basis of selecting the maximum value of this parameter and a sample upper bound value based on the existing FSAR information.

d. Main steam pressure and feedwater temperature

In Table 5-6, it is indicated that the maximum main pressure and temperatures corresponding to 100 percent power are used. Are there any technical specifications, initial condition ranges or trip set points used to determine the bounding values? Describe the method and the basis of selecting the maximum value of these two parameter and a sample upper bound value based on the existing FSAR information.

e. Reactor pool temperature

In Table 5-6, the nominal pool temperature is used. Although sensitivity calculations were documented in Section 9.6.5, a conservative value still needs to be selected based on technical specifications, initial condition ranges or trip set points. Describe the method and the basis of selecting the maximum value of this parameter and a sample upper bound value based on the existing FSAR information.

Staff recently drafted a request for additional information (RAI 9351, Question 3) regarding NRELAP5 steam generator model validation based on SIET tests. It appears that the model may overestimate the steam generator heat transfer capacity due to potential test data post processing errors. If that RAI results in steam generator model changes, the RAI responses for the above mentioned parameters need to reflect the updated information.