NuScaleTRRaisPEm Resource

From: Cranston, Gregory
Sent: Saturday, June 16, 2018 8:03 AM
To: NuScaleTRRaisPEm Resource
Cc: Lee, Samuel; Karas, Rebecca; Lu, Shanlai; Franovich, Rani; Chowdhury, Prosanta; Request for Additional Information
Subject: Request for Additional Information Letter No. 9519 (eRAI No. 9519) Topical Report, LOCA , 15.06.05, SRSB PUBLIC
Attachments: Request for Additional Information No. 9519 (eRAI No. 9519)-public.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: 93

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From: Cranston, Gregory

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Options
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Request for Additional Information No. 9519 (eRAI No. 9519)-public
Issue Date: 06/16/2018
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: 15.6.5

QUESTIONS

15.06.05-20

Title 10, Part 52, of the Code of Federal Regulations (10 CFR Part 52), "Licenses, Certifications, and Approvals for Nuclear Power Plants," Section 52.47, "Contents of Applications; Technical Information" (10 CFR 52.47), specifies that an application for certification of a nuclear power reactor design that uses simplified, inherent, passive, or other innovative means to accomplish its safety functions must meet the requirements of 10 CFR 50.43(e) (52 Part 52.47(c)(2)). 10 CFR 50.43(e) requires, in part, assessment of the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences. 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. 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 that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

As stated in RG 1.203, an evaluation model (EM) is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, as illustrated by the following examples:

(1) Procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation),
(2) Specification of those portions of the analysis not included in the computer programs for which alternative approaches are used, and
(3) All other information needed to specify the calculational procedure.

The entirety of an evaluation model (EM) ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM.

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