

NuScaleDCRaisPEm Resource

From: Cranston, Gregory
Sent: Friday, June 15, 2018 9:59 AM
To: Request for Additional Information
Cc: Lee, Samuel; Karas, Rebecca; Schmidt, Jeffrey; Franovich, Rani; Chowdhury, Prosanta; NuScaleDCRaisPEm Resource
Subject: Request for Additional Information No. 488 eRAI No. 9525 (15)
Attachments: Request for Additional Information No. 488 (eRAI No. 9525).pdf

Attached please find NRC staff's request for additional information (RAI) concerning review of the NuScale Design Certification Application. Password will be sent separately.

Please submit your technically correct and complete response within 60 days of the date of this RAI, or provide an alternate date within 14 days, 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. 488 (eRAI No. 9525)

Issue Date: 06/15/2018

Application Title: NuScale Standard Design Certification - 52-048

Operating Company: NuScale Power, LLC

Docket No. 52-048

Review Section: 15 - Introduction - Transient and Accident Analyses

Application Section:

QUESTIONS

15-28

10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, states:

"Under the provisions of 10 CFR 52.47, 52.79, 52.137, and 52.157, an application for a design certification, combined license, design approval, or manufacturing license, respectively, must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public."

Principle Design Criterion (PDC) 27 in FSAR, Tier 2, Section 3.1.3.8 states:

The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions, without margin for stuck rods...

The applicant evaluates the return to power scenario in FSAR, Tier 2, Section 15.0.6 to demonstrate that fuel integrity is maintained during a return to power event initiating from a design basis event (Anticipated Operational Occurrence [AOO] or Postulated Accident [PA]), assuming one control rod stuck out of the core, such that a safe stabilized condition is achieved and maintained during the event. Since the event can occur, using design basis assumptions, within a few hours of an AOO or PA, the event is considered to be within the design basis (see related RAI 9498).

Further, in FSAR, Tier 2, Section 8.3, the applicant references TR-0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical System." The NRC staff's safety

evaluation for TR-0815-16497 includes an evaluation of safe shutdown and established Condition 4.5 which requires an applicant referencing TR-0815-16497 to "demonstrate that the reactor can be brought to a safe shutdown using only safety-related equipment in the absence of electrical power following a DBE, with margin for stuck rods. Alternatively, an applicant addressing this condition may provide justification, for NRC review, for a less restrictive approach" (ML17340A524). The applicant addressed Condition 4.5 by requesting an exemption to General Design Criterion (GDC) 27, and proposing the less restrictive PDC 27.

In multiple sections of the NuScale Design Certification Application (DCA), including, but not limited to, the examples noted below, statements are made indicating that the plant achieves and maintains a safe shutdown condition following design basis events, which does not appear to be complete and accurate considering the exemption to GDC 27. While these statements would be accurate for situations where all rods successfully insert, or the core is in a part of the cycle when a return to power is not anticipated, the statements do not appear complete with respect to all design basis events (DBE). Some examples include the following (bold added for emphasis):

Section 15.0.6, "Evaluation of a Return to Power," states:

*For those events that rely on heat removal by the DHRS, heat produced after a return to power with a stuck control rod will be limited by negative moderator temperature feedback. The time to a return to power and the power level attained are based on conservative assumptions for the purpose of demonstrating fuel protection and **are not indications of plant shutdown capability.***

Section 15.0.0.6.3, "Engineered Safety Features Characteristics," states:

*The DHRS is designed to remove post-reactor trip residual and core decay heat from operating conditions and **transition the NPM to safe shutdown conditions without reliance on external power.***

Section 8.1.1, "Utility Power Grid and Offsite Power System Description," states:

*A loss of voltage, degraded voltage condition, or other electrical transient on the nonsafety-related AC power systems **has no adverse effect on the ability to achieve and maintain safe-shutdown conditions.***

The applicant is requested to update and clarify these and other similar statements for consistency throughout the FSAR to ensure they are complete with respect to the implications of the design basis event in Section 15.0.6 and PDC 27 in that the reactor

would achieve and maintain a safe, stabilized condition, but not necessarily a shutdown (subcritical) condition for all DBEs with the assumption that one rod does not successfully insert.