August 14, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 488 (eRAI No. 9525) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 488 (eRAI No. 9525)," dated June 15, 2018

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9525:

- 15-28

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9525

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9525

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9525
Date of RAI Issue: 06/15/2018

NRC Question No.: 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.

NuScale Response:

In the NuScale FSAR sections discussing the decay heat removal system (DHRS) (i.e., Final Safety Analysis Report (FSAR) Sections 15.0.6, 15.0.0.6 and 5.4) the term "Safe Shutdown" refers to the operating mode defined in Technical Specifications Table 1.1-1. The function of the DHRS is to remove residual and core decay heat following a reactor trip or normal plant shutdown. The term safe shutdown is used to describe the transition between operating modes and is not referring to reactivity control. Therefore, the usage in these sections is not related to the worst rod stuck out (WRSO) assumption.

As NRC has pointed out, the evaluation of a return to power with the WRSO assumption as described in Section 15.0.6 is part of the design basis of the NuScale Power Module (NPM). However, the capability of the NuScale design to achieve and maintain safe shutdown with all control rods is also part of the design basis of the NuScale design. The NPM design basis for a return to power with a WRSO and the NPM design basis for safe shutdown with all control rods inserted is described in Section 4.3.1.5 of the FSAR and in Section 15.2 of Part 7 of the DCA. FSAR Section 4.3.1.5 states:

> *For design basis events (DBE), the insertion of all CRAs provides the safety related means to shut down the reactor and maintain it in a shutdown condition. Long term shutdown capability is defined as the amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all CRAs are fully inserted and the RCS is cooled to equilibrium conditions. Long term shutdown capability is evaluated assuming that the core is xenon-free, no decay heat or voiding is present, and equilibrium samarium is accounted for. Insertion of all CRAs satisfies the portion of GDC 26 and PDC 27 requiring that one of the systems shall be capable of holding the reactor core subcritical under cold conditions.*

Per FSAR Section 4.3.1.5 and consistent with PDC 27, the NuScale design basis with respect to shutdown margin with an assumed WRSO establishes that core cooling is maintained through conservative safety analysis of DBE's, including the unlikely event of a return to power as analyzed in Section 15.0.6.

Per Section 4.3.1.5 and consistent with PDC 27, the NuScale design basis with respect to long term shutdown capability refers to the safety-related capability to provide safe shutdown for design basis events (DBE) through the insertion of all CRAs.

Further, per Section 15.2 of Part 7 of the DCA under the heading "NuScale Power Plant Design Basis and Safety with Respect to Intent of GDC 27."

> *The NuScale Power Module (NPM) design basis assures fuel protection under postulated accident conditions assuming WRSO and under conservative safety analysis methodology, but relies on all control rods inserted to assure long-term subcriticality under normal and accident conditions.*

Therefore, use of the term "safe shutdown" throughout the FSAR is either referring to the operational mode three as defined in Technical Specifications or the safety related reactivity control design basis of subcritical with all control rods inserted. Based on this position, the existing FSAR wording is considered appropriate.

Impact on DCA:

There are no impacts to the DCA as a result of this response.