RAIO-0718-60799

July 09, 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. 471 (eRAI No. 9496) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 471 (eRAI No. 9496)," dated May 10, 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. 9496:

- 15-19

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. 9496

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9496

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

eRAI No.: 9496
Date of RAI Issue: 05/10/2018

NRC Question No.: 15-19

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

> *Under the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. 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.*
>
> *These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.*

As the return to power analysis in the Final Safety Analysis Report (FSAR), Section 15.0.6, can occur, assuming a stuck rod, within a few hours from either an anticipated operational occurrence (AOO) or a postulated accident initiating event, the AOO acceptance criteria of GDC 10 apply. GDC 10, "Reactor design," requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs.

Primary Design Criterion (PDC) 27 in FSAR 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 provided the specified acceptable fuel design limits for critical heat flux would not be exceeded by the return to power.

The meaning of first paragraph is unclear with regard to a combined capability of reliably controlling reactivity when only one safety-related means of controlling reactivity is available. Further, the meaning of "specified acceptable design limits for critical heat flux would not be exceeded by the return to power" is given in the second paragraph, but is attributed to the situation where there is no stuck rod, when instead it should refer to the situation where there is a stuck rod. Please clearly separate those two situations such that the design criteria are consistent with the evaluated conditions.

Further, the staff notes that all fuel SAFDLs, of which minimum critical heat flux ration (MCHFR) is but one, should not be exceeded on a return to power consistent with GDC 10. Based on staff review of FSAR Section 15.0.6 it appears that only the MCHFR SAFDL has been evaluated during a return to power. The staff is requesting additional information regarding the evaluation of other SAFDLs, such as rod internal pressure criteria, to ensure that all relevant SAFDLs have been met consistent with GDC 10 and the wording of PDC 27 be modified to enhance its clarity.

The staff also notes that FSAR Section 3.1.3.8, Criterion 27 - Combined Reactivity Control Systems Capability, Implementation in the NuScale Power Plant, states:

The CVCS, with boron addition, and CRDS are designed for a combined capability of controlling reactivity changes that assures the capability to cool the core under postulated accident conditions with margin for stuck rods as explained in Section 4.3.1.5.

The staff notes that the NuScale chemical and volume control system (CVCS) is not credited to mitigate any Chapter 15 event and is unclear on the basis for including it to demonstrate the combined reactivity. The staff is requesting that the FSAR Section 3.1.3.8 clearly delinate the role of the CVCS in the NuScale in the design.

NuScale Response:

Item 1

Critical heat flux is identified as the most challenged specified acceptable fuel design limit (SAFDL) during the recriticality phenomenon as the event is similar to a slow decrease in heat removal anticipated operational ocurrence (AOO) event. Because of the non-limiting nature of the event, other SAFDLs were not considered relevant. This is described and justified in the

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exemption request, Part 7 Section 15.1.4 where the evaluation of other SAFDLs for other event types is considered to bound the return to power phenomenon. Specifically the bank withdrawal AOOs result in much more rapid and severe power excursions and the spurious RPV valve opening event causes a much more rapid temperature and pressure transient on the fuel. NuScale has determined that, consistent with GDC-10, the reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

Item 2

FSAR Section 4.3.1.5, *Shutdown Margin and Long Term Shutdown Capability*, explains that the chemical and volume control system (CVCS) is used to ensure allowable operating conditions are maintained, specifically minimum shutdown margin, such that the credited shutdown capability in the accident analysis is available at all times. No reactivity for addition of soluable boron is credited after the initiation of any Chapter 15 event. FSAR Section 4.3.1.5 states:

> *For postulated accidents comprised of infrequent events and accidents as described in Section 15.0, rapid CRA insertion after a reactor trip provides protection of the core. As with AOOs, the CVCS is used to adjust soluble boron concentration and maintain SDM prior to the event. Thus, for postulated accidents, the combined capability of the CVCS and CRAs control reactivity and ensures that the capability to cool the core is maintained as described in Section 15.0. CRAs reliably control reactivity changes after a postulated accident without the need for poison addition.*

FSAR Section 3.1.3.8, *Criterion 27-Combined Reactivity Control Systems Capability*, references the description in section 4.3.1.5 and, therefore, includes the appropriate level of detail with out further changes.

Impact on DCA:

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