March 12, 2019

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 Supplemental Response to NRC Request for Additional Information No. 26 (eRAI No. 8840) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 26 (eRAI No. 8840)," dated May 22, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 26 (eRAI No.8840)," dated July 19, 2017
3. NuScale Power, LLC Supplemental Response to NRC "Request for Additional Information No. 26 (eRAI No. 8840)," dated May 14, 2018
4. NuScale Power, LLC Supplemental Response to NRC "Request for Additional Information No. 26 (eRAI No. 8840)," dated July 2, 2018

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

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Question from NRC eRAI No. 8840:

- 19-2

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

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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8840

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8840

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

eRAI No.: 8840

Date of RAI Issue: 05/22/2017

NRC Question No.: 19-2

10 CFR 52.47(a)(27) states that a DC application must contain a Final Safety Analysis Report (FSAR) that includes a description of the design-specific probabilistic risk assessment (PRA) and its results. In accordance with the Statement of Consideration (72 FR 49387) for the revised 10 CFR Part 52, the staff reviews the information contained in the applicant's FSAR Chapter 19, and issues requests for additional information (RAI) and conducts audits of the complete PRA (e.g., models, analyses, data, and codes) to obtain clarifying information as needed. The staff uses guidance contained in Standard Review Plan (SRP) Chapter 19.0 Revision 3, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors." In accordance with SRP Chapter 19.0 Revision 3, the staff determines if "the PRA reasonably reflects the as-designed, as-built, and as-operated plant, and the PRA maintenance program will ensure that the PRA will continue to reflect the as-designed, as-built, and as- operated plant, consistent with its identified uses and applications."

The staff has reviewed the information in the FSAR and examined additional clarifying information from the audit of the complete PRA and determined that it needs additional information to confirm that the PRA reasonably reflects the as-designed plant. The containment isolation function supports the passive core cooling and heat removal key safety functions by ensuring sufficient coolant inventory in the reactor pressure vessel and the containment vessel. The staff notes that FSAR Table 19.1-6, "System Success Criteria per Event Tree Sequence," assumes that containment isolation is guaranteed to succeed except for the chemical and volume control system (CVCS) pipe breaks outside containment and the steam generator tube failure (SGTF). The containment isolation function is accordingly not questioned in any of the Level 1 event trees except for the CVCS pipe breaks outside containment and the SGTF.

To allow the staff to evaluate the Level 1 model and assumptions related to the containment isolation function, the staff requests the applicant to explain how the containment isolation function can be guaranteed to succeed in the Level 1 accident sequences. In your response, please provide the following:

a. Identify the potential scenarios (combinations of pathways, equipment failures and human failure events) that could lead to coolant inventory loss from the reactor pressure vessel to outside of the containment vessel.

b. For the scenarios identified in a), explain how the containment isolation function is accounted for in the Level 1 model, if this function is necessary to support any key safety functions (e.g., passive safety functions).

c. For the scenarios identified in a), if the containment isolation function is not necessary to support any key safety functions, please describe any relevant analyses used to support this conclusion. Describe any uncertainty analyses performed for these supporting analyses.

d. Augment FSAR Table 19.1-21, "Key Assumptions for the Level 1 Full Power Internal Events Probabilistic Risk Assessment," and/or Table 19.1-23, "Key Insights from Level 1 Full Power, Internal Events Evaluation," accordingly with a discussion of the dependency of the passive safety functions on the containment isolation function. Include a discussion of the safety-significance of the active backup systems for scenarios resulting in failure of containment isolation.

NuScale Response:

NuScale is supplementing its response to RAI No. 8840 (Question 19-2) originally provided in letter RAIO-0717-55003, dated July 19, 2017, and supplemented in letters RAIO-0518-59975 and RAIO-0718-60731, dated May 14, 2018 and July 02, 2018, respectively. This supplemental response is provided as a result of discussions with the NRC during a public meeting held on January 29, 2019. This supplemental response replaces the responses to Item c provided in the preceding NuScale letters; however, the FSAR changes associated with those responses remain valid.

c.) For chemical and volume control system (CVCS) line breaks outside of containment and steam generator tube failures (SGTFs) (as identified in Item a) with a continued loss of coolant (i.e., failure of containment isolation), core damage is avoided only if additional coolant via the

CVCS or the containment flooding and drain system (CFDS) is successful. Simulations demonstrate that coolant lost over 72 hours due to a containment isolation failure can be replenished by makeup coolant. For CVCS line breaks outside of containment and SGTFs, successful containment isolation retains sufficient coolant inventory in the NuScale Power Module (NPM) that makeup coolant is not required for success of the passive cooling functions of the decay heat removal system (DHRS) or emergency core cooling system (ECCS).

For loss of coolant accidents (LOCAs) inside containment, simulations have been performed to consider event tree model uncertainty. Simulations were performed including a failure of containment isolation on the containment evacuation system (CES) line penetration. The CES line is open during normal NPM operation to maintain sub-atmospheric conditions in the containment vessel (CNV); as such, it is the most likely containment bypass path despite the low probability of failure of both safety-related CES containment isolation valves (CIVs). Containment penetrations and their methods of isolation are listed in FSAR Table 19.1-24. While this scenario assumes the failure of the CES CIVs to close, the CES vacuum pumps would terminate operation and the suction and discharge valves to the pumps would close. The CNV would effectively remain a closed system unless the CES piping also fails. This is conservatively modeled in this sensitivity assessment by exposing the containment to an atmospheric pressure boundary condition through the 2-inch CES line for events in which the containment pressure exceeds the CES design pressure. Continued operation of the vacuum pumps in this beyond-design-basis event would require additional system failures that are independent of the failure to close the CES CIVs, and because the vacuum pumps are not designed to operate under the thermal hydraulic conditions of a LOCA inside containment.

- As indicated in FSAR Section 9.3.6.2.1, the CES is controlled by the nonsafety-related module control system (MCS) which provides both automatic and operator control of key CES functions, including valve alignment and vacuum pump operation. The suction and discharge side of the CES vacuum pumps are equipped with isolation valves with remote position indication and actuation, as well as pressure and temperature instruments for the purposes of performance monitoring and equipment protection in response to off-normal conditions. In the postulated LOCA sequence with failure to close CES CIVs, numerous off-normal signals would automatically trip the vacuum pump and close the suction and discharge valves. Additionally, the module protection system relays the safety-related containment isolation signal to the MCS as an additional trip on CES vacuum pump operation. Continued vacuum pump operation in the postulated LOCA conditions would require failure of the MCS in addition to and independent of the failure to close both CES CIVs.

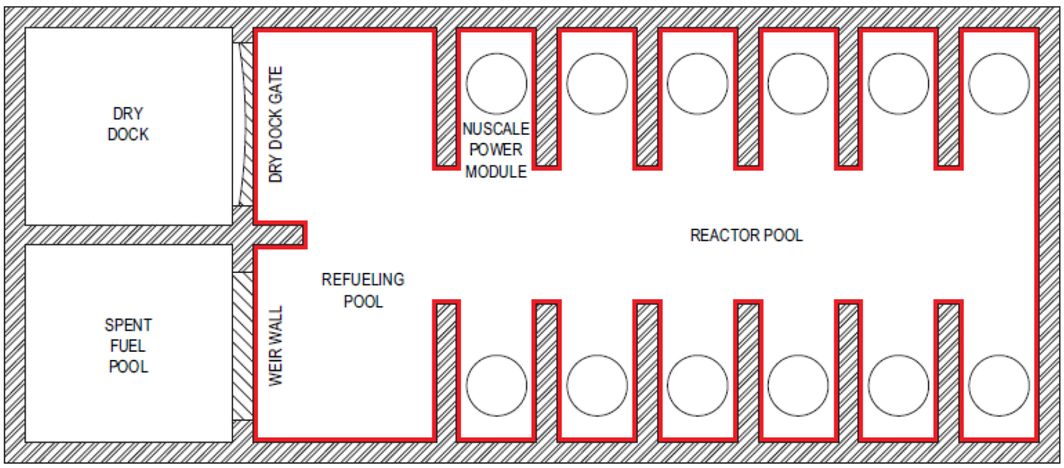
- Vacuum pumps are designed to operate between near vacuum and atmospheric suction pressures. Sensors, alarms, interlocks, and trips protect CES equipment from excessive conditions outside the design operating range. Irrespective of control signals, CES vacuum pumps are very unlikely to remain operable if exposed to the extreme pressure and temperature conditions in the CNV during the postulated LOCA sequence.

Simulation results demonstrate that one train of the ECCS with success of the reactor trip system (RTS) following a failure of CES isolation is sufficient to maintain the coolant levels in the reactor pressure vessel (RPV) and CNV to provide core cooling by circulation through the ECCS valves. The ECCS effectively depressurizes the RPV and the driving force for coolant loss from the CNV stops when the CNV pressure is reduced to atmospheric pressure. As a result, passive fuel cooling is provided for more than 72 hours. Note that in the very unlikely case in which both the RTS and automatic containment isolation fail following a LOCA inside containment, there is adequate time for operator action to either isolate the CNV or initiate makeup coolant using either the CVCS or CFDS to prevent core damage. The simulations were performed with the NRELAP5 code as identified in FSAR Section 19.1.4.1.1.6.

The temperature of the ultimate heat sink (UHS) affects the CNV pressure during a postulated accident sequence and in turn, the potential inventory loss that may be released through an open penetration. That is, cooler UHS temperatures maintain the CNV near atmospheric pressure while warmer UHS temperatures can result in higher CNV pressures and a relatively higher inventory loss rate. The NRELAP5 simulations indicate that the NPM remains at atmospheric pressure if the pool temperature remains at or below approximately 150 degrees Fahrenheit. Consequently, assumptions regarding the initial pool temperature, the heat load into the pool, and the pool volume are relevant in this assessment.

The NRELAP5 simulations assumed an initial reactor pool temperature of 100 degrees Fahrenheit, consistent with FSAR Section 9.2.5.2 which states that the UHS normal operating temperature is 100 degrees Fahrenheit. Although the maximum UHS temperature operating limit established in Technical Specification 3.5.3, Revision 2, was 140 degrees Fahrenheit, this limit is being reduced to 110 degrees Fahrenheit in Technical Specification 3.5.3, Revision 3, as stated in letter RAIO-0119-64318, which provides a response to RAI No. 9482 (Question 06.02.01.01.A-18). Based on engineering judgment, increasing the initial pool temperature in the NRELAP5 simulations to the revised Technical Specification operating temperature limit would have minimal effect on the progression of the scenario and would not change the conclusion of the NRELAP5 simulations.

The NRELAP5 analysis included a number of simplifying, conservative assumptions and did not credit active pool cooling systems, inventory addition, or ambient heat loss from the pool to the surrounding structures. In the NRELAP5 model, the UHS consists of the twelve module bays, the center channel, and the refueling pool, and is modeled as a well-mixed, bulk volume. The spent fuel pool (SFP) fluid volume and heat load are not included as part of the modeled UHS because of variable SFP heat loading, limited mixing with the rest of the pool, and because it is a net heat sink in some situations (that is, it is conservative to not include the combined SFP volume and heat load). A simplified schematic outlines the modeled UHS pool volume in the following figure.



In the “base” case simulation, a single NPM deposits heat into the modeled pool volume. A “limiting” case was also evaluated using the modeled pool volume divided by twelve (i.e., corresponding to one module bay and 1/12th of the central channel and refueling pool volumes). The limiting case results in the bulk UHS pool temperature increasing twelve times faster than the base case and approximates twelve NPMs depositing their decay heat load into the full pool. Even with this more challenging condition, the liquid level remains well above the core for the duration of the 72-hour mission time.

To evaluate the use of NRELAP5 with its well-mixed, bulk volume UHS model, an additional analysis was performed to independently model heat transfer and mixing in the UHS. Specifically, a computational fluid dynamics (CFD) model was used to analyze the flow of heated fluid in the UHS from a module bay to the volume of the refueling pool. The refueling pool was treated as a steady-state boundary condition of 100 degrees Fahrenheit. The best estimate, end-of-cycle decay heat from an NPM is less than 2.0 megawatts after approximately two hours and less than 1.0 megawatt after approximately 17 hours; the CFD analysis conservatively assumed a constant heat load of 2.0 megawatts. The CNV heat load boundary condition is representative of steady state ECCS operation and is independent of initiating event. The CFD analysis results demonstrate that effective mixing in the UHS occurs, as indicated by the insignificant local heatup in the module bay. The temperature in the module bay remains within three degrees of the bulk UHS temperature; this minor temperature difference supports the assumption in the NRELAP5 model that the UHS will be well-mixed, and hence it is realistic to represent the UHS as a bulk volume. Similarly, the CFD analysis supports the NRELAP5 result that the CNV remains near or below atmospheric pressure thereby minimizing the amount of coolant lost from the CNV. While the simulations were performed at a pool temperature of 100 degrees Fahrenheit, temperature-dependent properties of water including viscosity and thermal expansivity indicate that natural circulation flow and mixing will be enhanced at higher temperatures.

Impact on DCA:

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