UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 31, 2018

MEMORANDUM TO: Samuel S. Lee, Chief
Licensing Branch 1
Division of Licensing, Siting,
and Environmental Analysis
Office of New Reactors

FROM: Rani L. Franovich, Senior Project Manager ***/RA/***
Licensing Branch 1
Division of Licensing, Siting,
and Environmental Analysis
Office of New Reactors

SUBJECT: SUMMARY OF MAY 8, 2018, AND MAY 22, 2018, PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC, TO DISCUSS VARIOUS TOPICS RELATED TO CHAPTER 19, "PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION," OF THE NUSCALE DESIGN CERTIFICATION APPLICATION (DOCKET NO. 52-048)

On May 8, 2018, and May 22, 2018, representatives of the U.S. Nuclear Regulatory Commission (NRC) and NuScale Power, LLC (NuScale), held a public teleconference meeting. The purpose of these meetings was to discuss NuScale's responses to the NRC staff requests for additional information Nos. 9092, 9196, 9128, 8899, and 9112 related to Chapter 19.

A complete copy of NuScale's Design Certification Application is available on the NRC public Webpage at https://www.nrc.gov/reactors/new-reactors/design-cert/nuscale/documents.html.

Enclosure 1, "Summary of the May 8 and 22, 2018, Teleconference between the NRC staff and NuScale," provides a summary of the topics discussed during the teleconference.

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The agenda and list of meeting attendees are provided in Enclosures 2 and 3, respectively. The meeting notice is available in the NRC's Agencywide Documents Access and Management System, under Accession Nos. ML18110A394 and ML18117A330.

Docket No. 52-048

Enclosures:
1. Meeting Summary
2. Agenda
3. Attendees

SUBJECT: SUMMARY OF MAY 8 AND 22, 2018, PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC, TO DISCUSS VARIOUS TOPICS RELATED TO CHAPTER 19, “PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION,” OF THE NUSCALE DESIGN CERTIFICATION APPLICATION (DOCKET NO. 52-048)
DATED: July 31, 2018

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OFFICE	NRO/DLSE/LB1: PM	NRO/DLSE/LB1: LA	NRO/DSRA/SPRA: BC
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DATE	07/19/2018	07/16/2018	07/26/2018
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NAME	RKaras	SSamaddar	
DATE	07/31/2018	07/31/2018	

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U.S. NUCLEAR REGULATORY COMMISSION

SUMMARY OF MAY 8, 2018, AND MAY 22, 2018,

PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC

TOPIC 1: Chapter 15, "Transient and Accident Analyses"

NuScale Power, LLC's (NuScale) response to the following requests for additional information (RAI) :

RAI 9092 (Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition)

The U.S. Nuclear Regulatory Commission (NRC) staff discussed the following observations regarding NuScale's response:

- Regarding page 4 of the response: in the bulleted list under, "NuScale Response," the sequence of events table, not the key inputs table, should have been the first table for each event based on the attached markup and Final Safety Analysis Report (FSAR), Revision 1 (e.g., for the FSAR Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition," event, Table 15.4-25, "Sequence of Events for Limiting RCS Pressure Case (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition)," is the sequence of events, and Table 15.4-26, "Key Inputs for Limiting RCS Pressure Case (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition)," is the key inputs table; this is true for the FSAR Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," and Section 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error)," events as well). This constitutes only an error in the response to the RAI and not in the FSAR.

- Regarding page 13 of the response and FSAR markup showing Table 15.4-25, based on an audit of the calculation note for FSAR Section 15.4.1, the NRC staff recalls it was the high pressurizer pressure trip that happened first, and not the high power limit as the table appears to indicate.

- Regarding page 18 of the response and FSAR markup showing Table 15.4-30, "Key Inputs for Limiting RCS Pressure Case (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)": Footnote 2 appears to be in error, as the initial power level is 100 percent, not 75 percent.

The NRC staff noted that the FSAR markup provided in the RAI response has been incorporated into FSAR Revision 1.

- The updated timing in Table 15.4-1, "Sequence of Events for Limiting MCHR [minimum critical heat flux ration] Case (15.4.1 Uncontrolled CRA [Control Rod Assembly] Withdrawal from Subcritical or Low Power Condition)," in Revision 1 of the FSAR, appears to correspond to the limiting MCHFR case in NRELAP5, not VIPRE. As a result, the timing in Table 15.4-1 is not consistent with the time scales shown in the figures related to Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from Subcritical or Low Power

Condition." The timing in the table and the figures should be consistent or otherwise explained in the FSAR.

NuScale indicated that the RAI response will be supplemented to correct these errors.

TOPIC 2: Chapter 19, "Probabilistic Risk Assessment And Severe Accident Evaluation"

NuScale response to the following RAIs:

RAI 8899, Q19.01-16 (Seismic assumptions for the Containment Flange Tool (CFT) and Refueling Flange Tool (RFT))

In response to the NRC staff talking point, NuScale will submit a supplemental RAI response with the basis for the key assumption regarding the CFT and RFT remaining connected to the Reactor Building Crane in Table 19.1-40: Key Assumptions for the Seismic Margin Assessment as "expected operating practice."

RAI 9196, Q19-36 (interfacing system loss of coolant accidents, ISLOCAs)

SECY 93-87, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs on ISLOCA," states,

> *The staff further recommended that systems that have not been designed to full RCS pressure should include:*
>
> - *the capability for leak testing of the pressure isolation valves; valve position indication that is available in the control room when isolation valve operators are deenergized;*
> - *and, high-pressure alarms to warn control room operators when rising reactor coolant (RC) pressure approaches the design pressure of attached low pressure systems and both isolation valves are not closed*.

SECY 93-087 also states,

> *the degree of isolation or number of barriers (for example, three isolation valves) is not sufficient justification for using low-pressure components that can practically be designed to the full RCS URS (ultimate rupture strength) criterion.*

The NRC staff acknowledges that ISLOCAs are mitigated by actuation of the Chemical and Volume Control System (CVCS) containment isolation valves on reactor coolant system (RCS) pressure and RCS level signals (FSAR Table 7.1-4). The NRC staff also understands that CVCS containment isolation valves have position indication provided in the control room, and they have the capability for leak testing and are periodically leak tested as part of the Inservice Testing Program (FSAR Section 3.9.6).

Regarding high pressure alarms on attached low pressure systems, such as the CVCS makeup line upstream of the CVCS makeup pump and the purification bypass line (as depicted on Figure 9.3.4-1: Chemical and Volume Control System Diagram), the RAI response states the CVCS makeup pumps have integral check valves, and a check valve is provided between the makeup pumps' discharge and the injection tee into the recirculation loop. However, the RAI response and FSAR update did not provide information on how RCS leakage/high pressure is being monitored in these low pressure makeup lines.

In response to the NRC staff talking points, NuScale confirmed that the CVCS purification bypass line is designed to RCS full pressure. Regarding the CVCS makeup line upstream of the CVCS makeup pump (as depicted on FSAR Figure 9.3.4-1: Chemical and Volume Control System Diagram), NuScale will provide a supplemental RAI response with updates to FSAR Figure 9.3.4-1. This revised figure will show the pressure transmitters on the suction side and discharge side of the makeup pump. This revised figure will also show high pressure alarm on the suction side of the makeup pump. The revised figure will also show that the purification bypass line is designed to full RCS pressure. During the meeting, the NRC staff reiterated the position in SECY-93-087 that systems connected to the RCS (including CVCS subsystems and CVCS connecting systems) be able to withstand full RCS pressure or be instrumented to identify leakage past isolation valves separating high pressure piping from low pressure piping. The supplemental RAI response will confirm that the search for low pressure piping in systems connected to the RCS is complete.

<u>RAI 9127, Q19-37 (module drop events during module movement)</u>

The NRC staff discussed the following observations with NuScale:

- The difference between their module drop probabilities and the previous heavy load drop estimates developed in EPRI Report 1009691, “Probabilistic Risk Assessment (PRA), of Bolted Storage Casks, Revised Quantification and Analysis Report,” published in December 2004, and NUREG-1774, “A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002,” published in July 2003.

- The need for risk-significant operator actions pertaining to module drop events need to be documented in Chapter 19 of the DCD so that the human factors reviewers can consider these operator actions during their review (Chapter 18 of the DCD), in addition to control room actions.

The NRC staff discussed the difficulty in reconciling NuScale’s module drop probability and previous heavy load drop estimated developed in Electric Power Research Institute (EPRI) Report 1009691, “Probabilistic Risk Assessment (PRA) of Bolted Storage Casks: Updated Quantification and Analysis,” and NUREG-1774 based on the RAI response, FSAR, Revision 1, Chapter 9, and FSAR, Revision 1, Chapter 19. The NRC staff explained that NuScale’s failure probability per lift is two orders of magnitude lower than estimated in EPRI Report 1009691, and three orders of magnitude lower than estimated in NUREG-1774.

The NRC staff re-iterated that based on the applicant’s Reactor Building Crane PRA, the EPRI Report, and NUREG-1774, operator errors are significant contributors to module drop, similar to other shutdown evolutions in current operating reactors such as reactor vessel drain down in PWRs. The NRC staff and applicant discussed NuScale’s crane design improvements from the EPRI report and NUREG-1774. NuScale noted that the reactor building crane design has evolved since the risk analysis of the crane was completed, and additional design improvements are being considered. Regarding the EPRI risk results, NuScale’s reactor building crane has redundant gear boxes. Regarding the NUREG-1774 risk results, NuScale does not use temporary riggings for the module, and there are double sets of wires. The NRC staff learned that NUREG-7016, “Human Reliability Analysis-Informed Insights on Cask Drops,” published February 2012, was not used to inform NuScale’s drop module analysis.   In the NuScale design, limit switches and instrumentation detect and prevent operator errors during module movement. NuScale indicated that Chapter 9 does not contain much detail on the limit switches.

The applicant agreed to supplement the RAI response to provide the justification for the reduction in drop probabilities. NuScale agreed to update Table 19.1-71, “Low Power and Shutdown PRA Key Assumptions,” with the safety features of the crane that need to be available during module movement. NuScale will evaluate Table 17.4-1 to ensure that the link from the reactor building crane and module lift adapter indication and instrumentation to the Maintenance Rule is clear.

The NRC staff also observed that the frequency of module movement at the facility is one module refueling evolution every two months and discussed how these insights are evaluated in the multi-module risk discussion, noting that the frequency of module movement and postulated module drops was not evaluated in the multi-module analysis quantitatively. The NRC staff acknowledged that although module drop core damage events dominate the NuScale core damage frequency, they do not dominate the large release frequency, since module drop events do not result in a large release. The NRC staff is confirming this assumption via related RAIs. No further action from NuScale is needed.

RAI 9112, Q19-36 (high pressure melt ejection)

The NRC staff agreed that, in the event of a loss-of-coolant accident (LOCA) leading to core damage, the reactor pressure vessel (RPV) would depressurize by the time of RPV lower plenum dryout as shown by the analysis in FSAR Chapter 19.2. The NRC staff’s remaining issue is that there could be a re-pressurization of the RPV if operators add water to the RPV after RPV lower plenum dryout. The NRC staff asked whether late water addition could lead to a high pressure melt ejection. The NRC staff asked whether SAMG guidance has been or would be developed to warn the operators against late water addition.

NuScale stated that there is no need to constrain the operators regarding late water addition. NuScale stated that it was prepared to discuss material in the Electronic Reading Room that shows there is no need to constrain the operators regarding late water addition.

The NRC staff stated that a key staff member was on travel this week. NuScale suggested and the NRC staff agreed that the staff would review the material in the ERR and let them know whether another meeting with the staff is needed.

RAI 8840, Q19-2 (Containment isolation for LOCAs inside containment)

The NRC staff requested NuScale to address additional technical issues related to the refined N-RELAP thermal-hydraulic analysis that was performed to demonstrate that failed containment isolation is a success path (i.e., core damage is prevented) in the NuScale PRA. A part of NuScale’s technical basis was the computational fluid dynamics (CFD) analysis used to characterize the thermal mixing of the ultimate heat sink (UHS) when the reactor module is rejecting heat to the UHS.

Regarding the initial UHS temperature, NuScale stated that 40 degrees Farenheit was used based on a limiting design basis calculation and indicated that the mixing results in terms of the change in UHS temperature would be roughly the same if other conditions remained constant. NuScale stated that the CFD insights show that rejecting heat to only the local area surrounding the module is not realistic.

NuScale clarified that the postulated transient scenario for the heat load to the UHS was a CVCS discharge line break inside containment.

NuScale clarified that potential effect of multi-module trip (e.g., loss of AC power) was considered. NuScale stated that an analysis crediting the NPM bay, the reactor pool center channel, the refueling pool, and the spent fuel pool showed that core damage is prevented.

NuScale clarified that impact of spent fuel heat load was included in the evaluation.

NuScale stated that the NRELAP vertical nodalization scheme was changed from 10 vertical nodes to two to simulate a more realistic UHS temperature response. NuScale stated that explicit benchmarking of the code has not be performed for this type of analysis.

NuScale agreed to supplement the response to RAI 8840 to provide a explain how the above topics were considered to show that failed containment isolation is a success in the NuScale PRA.

U.S. NUCLEAR REGULATORY COMMISSION

SUMMARY OF MAY 8, 2018, AND MAY 22, 2018,

PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC

MEETING AGENDA

Tuesday, May 8, 2018

Time	Topic	Speaker
1:00 pm – 2:00 pm	NuScale Response to eRAI 9092	NRC/NuScale
2:00 pm – 4:00 pm	NuScale Response to eRAIs 9196, 9128, 8899 and 9112	NRC/NuScale

Tuesday, May 22, 2018

Time	Topic	Speaker
2:00 pm – 3:00 pm	NuScale Response to eRAI 8840	NRC/NuScale

Enclosure 2

LIST OF ATTENDEES

NuScale	**May 8, 2018**	**May 22, 2018**
	B. Bristol	G. Buster
	S. Bristol	J. Curry
	G. Buster	B. Galyean
	A. Callaway	C. Williams
	R. Charoensombud	
	J. Curry	
	B. Galyean	
	B. Haley	
	P. Infanger	
	M. McCloskey	
	L. McSweeny	
	D. Peebles	
	F. Perdomo	
	C. Williams	
NRC Staff	**May 8, 2018**	**May 22, 2018**
	A. Burja	O. Ayegbusi
	R. Franovich	C. Boyd
	A. Neuhausen	R. Franovich
	M. Pohida	T. Nakanishi
	J. Schaperow	C. Thurston
Public		
	None	None