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

May 24, 2018

MEMORANDUM TO: Samuel S. Lee, Chief
Licensing Branch 1
Division of New Reactor Licensing
Office of New Reactors

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

SUBJECT: SUMMARY OF FEBRUARY 13, 20, AND 27, 2018, PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC, TO DISCUSS VARIOUS TOPICS RELATED TO CHAPTER 19, "PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION," AND CHAPTER 15, "TRANSIENT AND ACCIDENT ANALYSES," OF THE NUSCALE DESIGN CERTIFICATION APPLICATION (DOCKET NO. 52-048)

On February 13, 20, and 27, 2018, representatives of the U.S. Nuclear Regulatory Commission (NRC) and NuScale Power, LLC, held a public teleconference meeting. The purpose of this meeting was to discuss the following:

1. NuScale's responses to the NRC staff Request for Additional Information (RAI) Nos. 8774 and 9205 related to Chapter 15.

2. NuScale's responses to the NRC staff RAI Nos. 8899, 8813, 8840, 8892, 9178, and 9068 related to Chapter 19.

A complete copy of NuScale's DCA 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 February 13, 20 and 27, 2018, Teleconference between the NRC staff and NuScale," provides a summary of the topics discussed during the teleconference.

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301-415-7334

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. ML18003A631, ML18003A635, and ML18003A635.

Docket No. 52-048

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

SUBJECT: SUMMARY OF FEBRUARY 13, 20, AND 27, 2018, PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC, TO DISCUSS VARIOUS TOPICS RELATED TO CHAPTER 19, "PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION," AND CHAPTER 15, "TRANSIENT AND ACCIDENT ANALYSES," OF THE NUSCALE DESIGN CERTIFICATION APPLICATION (DOCKET NO. 52-048) DATED: MAY 24, 2018

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OFFICE	NRO/DNRL/LB1: PM	NRO/DNRL/LB1: LA	NRO/DSRA/SPRA: BC
NAME	RFranovich	MMoore	MHayes*
DATE	5/24/2018	5/02/2018	5/2/2018
OFFICE	NRO/DSRA/SRSB: BC	NRO/DEI/MEB: BC	
NAME	RKaras (RNolan* for)	TLupold*	
DATE	5/23/2018	5/2/2018	

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U.S. NUCLEAR REGULATORY COMMISSION

SUMMARY OF FEBRUARY 13, 20, AND 27, 2018,

PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC

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

RAI 8744 (Crediting of nonsafety-related valves)

The U.S. Nuclear Regulatory Commission (NRC) staff proposed a path forward to credit nonsafety-related valves for Chapter 15, "Transient and Accident Analyses," event mitigation (feedline break, main steamline break, steam generator tube rupture (SGTR) and feedwater containment pressure analysis) and discussed the proposed approach by NuScale Power, LLC (NuScale) to follow the guidance in NUREG-0138, "Staff discussion of 15 Technical Issues Listed in Attachment to Nivember 3, 1976, Memorandum from Director, NRR to NRC Staff," to rely on nonsafety-related valves for specific applications in the NuScale reactor accident analysis. The three valves in questions are: 1) the nonsafety-related feedwater check valves, 2) nonsafety-related secondary (backup) main steam isolation valves (MSIVs), and 3) the nonsafety-related feedwater regulatory valve. The NRC staff indicated that this approach would be acceptable provided that NuScale satisfies the guidance in NUREG-0138 consistent with the accepted application at operating nuclear power plants. In particular, NuScale will need to demonstrate that the nonsafety-related valves to be credited in the accident analysis are reliably capable of performing their intended functions. For example, NuScale should (1) identify the specific valves, (2) indicate the valve type, (3) describe the performance history and operating experience with those valves and their application, (4) specify the design and qualification requirements to be applied to those valves, (5) specify the preservice and inservice testing, and technical specification (TS), requirements to be applied to those valves, (6) describe the planned modifications to the final safety analysis report (FSAR) and TSs to incorporate those requirements, and (7) clarify the augmented provisions indicated in FSAR Tier 2, Table 3.2-1, "Classification of Structures, Systems, and Components," for these valves.

The NRC staff tentatively agreed to allow event mitigation using nonsafety-related valves assuming NuScale agrees to meet initial qualification and inservice test requirements and performs an analysis to demonstrate accident dose limits are met assuming the secondary MSIV fails to close for the SGTR event. The dose analysis would demonstrate that the consequences of the tube rupture event are consistent with NUREG-0138, which allows for design basis event mitigation with nonsafety-related structure, system, and components. The staff stated that a follow-up RAI will be prepared to request this information. The NuScale participants did not have any questions regarding this discussion.

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

NuScale Power, LLC's Response to the following RAIs:

RAI 8899, Q19.01-8 (probabilistic risk assessment [Probabilistic Risk Assessment (PRA)]-based Seismic Margins Analysis [SMA])

The NRC staff requested NuScale to explain how the methodology typically used for a seismic

PRA, binning of the seismic hazard into 14 seismic initiating event trees, is used to support the PRA-based SMA. NuScale provided additional background on their use of the methodology. The NRC staff also requested NuScale clarify how "this methodology supports site-specific estimates of seismic hazard occurrence frequency." NuScale agreed to supplement this RAI response with an FSAR update that removes this statement.

RAI 8899, Q19.01-16 (SMA assumption)

The NRC staff requested NuScale to clarify the assumption that the module will remain attached to, and supported by, the reactor building crane until to the upper module is moved to dry dock. The NRC staff also requested NuScale that this assumption be described in FSAR Table 19.1-40, "Key Assumptions for the Seismic Margin Assessment." NuScale agreed to supplement the RAI response with a FSAR update that includes the clarified assumption in FSAR Table 19.1-40.

RAI 8813 (PRA emergency core cooling system [ECCS] model)

The NRC staff requested NuScale to clarify the basis for not explicitly modeling the inadvertent actuation block in the PRA for scenarios that may require the component to change state. NuScale agreed to supplement the RAI response with a FSAR update that includes the basis for screening out this emergency core cooling failure mode.

RAI 8840 (Containment isolation for loss-of-coolant accidents inside containment)

The NRC staff requested NuScale to clarify the basis for screening out loss-of-coolant accidents inside containment that occur in conjunction with containment isolation failure. Topics discussed included decay heat removal system performance and consideration of anticipated transients without scram for scenarios of concern. The NRC staff learned that consideration of passive system success concurrent with containment isolation failure was not evaluated in the Passive System Reliability Report. To facilitate resolution, the NRC staff requested that additional related technical information be made available for an upcoming audit. The NRC staff also requested NuScale to address the validity of certain proposed FSAR statements. NuScale agreed to propose appropriate FSAR revisions in a supplemental RAI response after the completion of the audit, along with any additional items from the audit.

RAI 8892 (Internal flooding assumption)

The NRC staff requested NuScale to describe the internal flooding assumption related to operator's ability to isolate any internal flood in the reactor building as a key assumption in FSAR Table 19.1-54. NuScale agreed to supplement the RAI response with a FSAR update that describes this key assumption in FSAR Table 19.1-54.

RAI 9178 (Internal fire assumption)

The NRC staff requested NuScale to clarify how fire-induced ground faults that may operate the ECCS valve solenoids are addressed in the risk assessment. NuScale provided the requested clarification and no further action is necessary.

RAI 9068 (Operator action assumptions)

The NRC staff requested NuScale to clarify the available cues operators will rely on to manually unisolate containment and initiate flooding from the containment flooding and drain system and chemical and volume control system (CVCS) versus the cues operators that will rely on to manually actuate containmnent isolation valves following the failure of the MPS to automatically isolate containment. The NRC staff also requested that NuScale discuss the viability of operators to align the demineralized water system (DWS) as a source of reactor coolant makeup considering potential fire and flood impacts on local manual actions. NuScale provided the requested clarification and no further action is necessary.

RAI 9068 (Human error probability)

The response to this RAI was discussed on February 27, 2018. The response to Question 19-28 states, “Thus, several hours are available for operators to locally unisolate and align the demineralized water system (DWS), if additional inventory is needed.” The proposed footnote to Table 19.1-14, “Modeled Human Actions,” does not specify that aligning DWS supply isolation valves is a local action. The NRC staff requested NuScale to update the footnote in Table 19.1-14 to specify that re-aligning the DWS supply valves is a local action. NuScale plans to submit a revised RAI response to RAI 9068, Question 19-28, which will include updates to Table 19.1-14 to specify that aligning DWS is a local action. NuScale also plans to update Table 19.1-15, “Generic Sources of Level 1 Model Uncertainty,” to include DWS system dependency to CVCS.

The NRC staff understands that the human error probability of DWS re-alignment was considered within the scope of CVCS injection. The NRC staff requested a clarification whether the boric acid storage tank level and batch tank level are monitored in the control room and are alarmed on low level. NuScale confirmed that the boric acid storage tank level can be monitored locally and in the control room and is alarmed on low level. In the revised RAI response to Question 19-28, NuScale plans to update the FASR either in Chapter 9, “Auxiliary Systems,” or Chapter 19 in the PRA assumption tables with this information. NuScale confirmed that random failures of DWS re-alignment were modeled in the PRA.

U.S. NUCLEAR REGULATORY COMMISSION

SUMMARY OF FEBRUARY 13, 20, AND 27, 2018,

PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC

MEETING AGENDA

Tuesday, February 13, 2018

Time	Topic	Speaker
1:00 pm – 5:00 pm	NuScale Response to eRAIs 8899, 8813, 8840, 8892, 9178, and 9068	NRC/NuScale

Tuesday, February 20, 2018

Time	Topic	Speaker
3:00 pm – 4:00 pm	NuScale Response to eRAIs 8774 and 9205	NRC/NuScale

Tuesday, February 27, 2018

Time	Topic	Speaker
1:00 pm – 3:00 pm	NuScale Response to eRAI 9068, Question 19-28	NRC/NuScale

LIST OF ATTENDEES

NuScale	February 13, 2018	February 20, 2018	February 27, 2018
	S. Bristol	B. Bristol	S. Bristol
	J. Curry	S. Bristol	J. Curry
	B. Galyean	E. Coryell	B. Galyean
	D. Gardner	D. Gardner	C. Williams
	B. Haley	P. Infanger	
	L. McSweeny	G. McGee	
	C. Williams	G. Myers	
		D. Peebles	

NRC Staff	February 13, 2018	February 20, 2018	February 27, 2018
	B. Bavol	C. Ashley	M. Pohida
	M. Caruso	R. Franovich	
	T. Nakanishi	S. Haider	
	A. Neuhausen	R. Nolan	
	M. Pohida	T. Scarbrough	
	B. Travis	J. Schmidt	
		M. Thomas	
		B. Travis	

Public			
	S. Fields	None	None