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

October 3, 2019

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 THE JANUARY 8, 2019, JANUARY 29, 2019, AND APRIL 17, 2019, PUBLIC TELECONFERENCES WITH NUSCALE POWER, LLC, TO DISCUSS THE REQUESTS FOR ADDITIONAL INFORMATION IN CHAPTER 15, "TRANSIENT AND ACCIDENT ANALYSES," AND CHAPTER 19, "PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION," OF THE NUSCALE DESIGN CERTIFICATION APPLICATION (DOCKET NO. 52-048)

On January 8, 2019, January 29, 2019, and April 17, 2019, representatives of the U.S. Nuclear Regulatory Commission (NRC) and NuScale Power, LLC (NuScale), held a public teleconference meeting. The purpose of the meeting was to discuss the NRC staff's Requests for Additional Information (RAI) Nos. 9512, 9504, 9507, 9483, and 9491, related to Final Safety Analysis Report (FSAR) Chapter 15, "Transient and Accident Analyses"; RAI 8840, related to FSAR Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation"; and the wording of FSAR Section 19.2, "Severe Accident Evaluation."

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 January 8, 2019, January 29, 2019, and April 17, 2019, 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 notices are available in the NRC's Agencywide Documents Access and Management System, under Accession Nos. ML19007A046, ML19002A196, and ML19067A041.

Docket No. 52-048

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

SUBJECT: SUMMARY OF THE JANUARY 8, 2019, JANUARY 29, 2019, AND APRIL 17, 2019, PUBLIC TELECONFERENCES WITH NUSCALE POWER, LLC, TO DISCUSS THE REQUESTS FOR ADDITIONAL INFORMATION IN CHAPTER 15, “TRANSIENT AND ACCIDENT ANALYSES,” AND CHAPTER 19, “PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION,” OF THE NUSCALE DESIGN CERTIFICATION APPLICATION (DOCKET NO. 52-048)
DATED: OCTOBER 3, 2019

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U.S. NUCLEAR REGULATORY COMMISSION

SUMMARY OF JANUARY 8, 2019, JANUARY 29, 2019, AND APRIL 17, 2019

PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC

Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation

Section 19.2, "Severe Accident Evaluation"

On January 8, 2019, U.S. Nuclear Regulatory Commission (NRC) staff and NuScale Power, LLC's (NuScale) continued to discuss the language in Final Safety Analysis Report (FSAR) Section 19.2, "Severe Accident Evaluation." This discussion began November 13, 2018 (ML18331A062). The NRC staff reiterated its concern with NuScale assertions that the failure of the reactor pressure vessel (RPV) lower head is not "physically realistic," an ex-vessel steam explosion is "physically unrealistic," and failure of the containment lower head is not "physically realistic." In response, NuScale offered the following:

1. NuScale proposed to replace "physically unrealistic" with "cannot occur" or "does not occur" because its analysis led to this outcome. The NRC staff asked NuScale to consider replacing absolute terms such as "physically unrealistic" and "cannot occur" or "does not occur" with probabilistic terms such as "unlikely" to reflect the phenomenological uncertainties identified by the staff. NuScale indicated it was disinclined to characterize failure of the RPV lower head as "unlikely" because the basis of its assertion was analytical, not probabilistic. The NRC staff also suggested the NuScale remove the word "conservative" from several passages in the FSAR because of phenomenological uncertainties not quantified in the analysis. NuScale indicated it would consider removing the word "conservative."

2. NuScale considered revising the FSAR to acknowledge phenomenological uncertainties (critical heat flux, focusing effect, and intermetallic reactions) identified by the NRC staff and proposed to add language on page 19.2-20 of the FSAR to address "uncertainties" and an example of uncertainty such as critical heat flux.

3. With respect to an alternative acceptance criterion (i.e., that the NuScale design is unlikely to lead to a large release following a severe accident) for containment performance during severe accidents involving ex-vessel steam explosion, NuScale indicated its intent to revise the FSAR to align with the alternative acceptance criterion. NuScale proposed to state that an ex-vessel steam explosion causing containment failure is unlikely to lead to a large release and to make conforming changes to reflect this statement in the FSAR discussion on ex-vessel steam explosion, including Table 19.1-32 on page 19.1-187. NuScale asked what metric the NRC staff is using to determine if a postulated release is "large." The NRC staff responded it was using the 2.9 percent iodine release metric, which is NuScale's large release metric for at-power accidents in FSAR Section 19.1.

4. With respect to in-vessel steam explosion and the NRC staff's confirmatory analysis, NuScale proposed to revise the FSAR to indicate its analysis assumed molten debris. The NRC staff responded that the proposed change partially addresses this issue but falls short of acknowledging additional phenomenological uncertainty, including the

possibility that corium temperatures could be higher than what was assumed in NuScale's probabilistic analysis for in-vessel steam explosion. NuScale indicated it understood the NRC staff's concerns and proposed to revise the FSAR and replace "assuming molten debris" with "assuming that pre-mixing was equivalent to that of molten debris, even though the debris was not hot enough to be molten."

Subsequent to the January 8, 2019, public meeting, and in response to issues raised therein, NuScale provided a mark-up of FSAR Chapter 19 via email on January 18, 2019.

On January 29, 2019, the NRC staff and NuScale met again to discuss the revisions to the language in FSAR Chapter 19. The NRC staff indicated that the FSAR Chapter 19 mark-up resolved the NRC staff's issues. In addition, the NRC staff proposed two clarifying changes to the FSAR Chapter 19 mark-up.

1. To delete the following passage in FSAR Section 19.1:

   > *These phenomena are not included in the [containment event tree (CET)] consistent with the approach taken in NUREG-1524 (Reference 19.1-64) in which phenomena that are judged to be 'physically impossible', 'vanishingly small,' or 'very unlikely' have a probability of less than 1E-3. Because this probability is small with respect to the large release frequency (LRF) and the conditional containment failure probability (CCFP) safety goal of less than 0.1, such events are not explicitly included in the CET.*

   The NRC staff expressed concern with this passage in light of phenomenological uncertainties, which are addressed in FSAR Section 19.2. NuScale agreed to consider revising this passage and repeated its assertion that the containment vessel is sufficiently robust to withstand severe accident challenges. The NRC staff responded that it did not believe this deletion would detract from NuScale's assertions.

2. To retain the following original language in FSAR Section 19.2 regarding hydrogen generation and control: "In summary, over-pressurizing the NuScale containment vessel due to hydrogen combustion is physically unrealistic due to the very limited oxygen concentration before and after postulated severe accidents." The NRC staff indicated it believed the original language more clearly described NuScale's hydrogen combustion analysis results. The phenomenological uncertainties that the NRC staff previously identified were associated with in-vessel retention and steam explosion and not with hydrogen combustion. NuScale agreed to consider restoring the original language.

<u>Request for Additional Information No. 8840</u>

In its response to Request for Additional Information (RAI) No. 8840, NuScale assumed that containment isolation is not required during various loss-of-coolant-accident (LOCA) events inside containment. This assumption is not reflected in the FSAR. The plant design is for containment isolation to actuate when there is a LOCA type event in containment as described in parts of the FSAR. However, since this is not modeled in the probalistic risk assessment, while potentially acceptable, the NRC staff believes the assumption should be captured in the FSAR (likely in the key assumptions table in Chapter 19).

On April 17, 2019, the NRC staff asked the following three questions about NuScale's response to RAI No. 8840:

1. Did the evaluation consider operation of the Containment Evacuation System (CES) vacuum pumps continuing to take a suction on containment during LOCA inside containment events?

   Response: There is no discussion in the FSAR about the CES system shutting down. NuScale responded that its evaluation did not consider the CES vacuum pumps and that the running pumps would not impact the evaluation. NuScale agreed to submit a supplemental response to the RAI and describe the impact of the CES vacuum pumps on containment during a LOCA inside containment scenario.

2. Why was 100 degrees Fahrenheit (°F) assumed rather than 140 °F?

   Response: The response states the simulation assumed a normal pool operating temperature of 100 °F. However, Technical Specification (TS) 3.5.3, Revision 2, states that the pool should be maintained at less than 140°F.

   Does the analysis change significantly when pool temperature is assumed to be 140 °F?

   Response: NuScale responded that the pool's TS limit has been reduced from 140 °F to 110 °F and agreed to provide the rationale for assuming a pool temperature of 100 °F for its evaluation in the supplemental response to the RAI.

3. How much of the pool was assumed in the simulation that was referred to in the July 2018 RAI response?

   Response: NuScale responded that the simulation assumed the LOCA event involving one module concurrent with decay heat load from the other shutdown modules. The simulation assumed the entire pool excluding the spent fuel pool was available to provide cooling. NuScale added that it had performed a sensitivity analysis on the evaluation, which assumed only one twelfth of the pool was available and showed no core damage in the first 72 hours for the module experiencing the LOCA event.

Chapter 15, "Accident and Transient Analyses"

On January 29, 2019, the NRC staff and NuScale discussed the following RAIs related to Chapter 15.

<u>RAI 9512, Question 15.04.03-2</u>

The markup of Table 15.4-33, "Key Inputs for Key Inputs for Limiting Control Rod Assembly Misalignment," shows that the nominal core inlet temperature of 487.4 °F, which corresponds to a low-biased RCS average temperature of 535 °F, is biased high by 10 °F. It is not clear to the NRC staff that the core inlet temperature is biased in a limiting manner, as it appears the net effect of the "nominal" (biased low) temperature plus the 10 °F bias corresponds to the nominal RCS average temperature of 545 °F. In other words, the combination of the nominal (biased-low) temperature and the applied bias appear to cancel out. Furthermore, the NRC staff notes that a biased-high RCS average temperature is typically limiting for minimum critical heat flux ratio (MCHFR). The NRC staff requested NuScale to explain the logic for the choice of the

nominal and biased core inlet temperature values, including why the nominal core inlet temperature does not correspond to the nominal RCS average temperature of 545 °F.

The NuScale responded that the control rod assembly misalignment analysis is a steady-state analysis that requires no interface with the transient code (NRELAP5). It is evaluated using the subchannel analysis methodology (VIPRE-01) with low RCS flow bias conditions, including the core inlet temperature corresponding to minimum RCS flow. Because the NuScale design is based on natural circulation, core inlet temperature and RCS flow are coupled for a given power level. In other words, the "nominal" 487.4 °F core inlet temperature corresponds to the normal core inlet temperature when RCS flow rate is biased low. Simultaneously assuming a core inlet temperature corresponding to nominal RCS flow and biasing RCS flow low would be unphysical because of the relationship between RCS flow and RCS temperature. The applicant also noted that differences exist in the treatment of RCS temperature between VIPRE-01 and NRELAP5 because RCS temperature inputs are specified differently in the two codes. NRELAP5 is based on an average RCS temperature control scheme, whereas the core inlet temperature is specified directly as a boundary condition in VIPRE-01.

RAI 9504, Question 15.04.07-3

The response to this RAI similarly states that the core inlet temperature bias corresponds to the minimum RCS average temperature range specified in FSAR Table 15.0-6. The NRC staff asked why a minimum RCS average temperature is used when a high bias is typically limiting for MCHFR. NuScale reiterated that the control rod assembly misalignment analysis is a steady-state analysis that requires no interface with the transient code (NRELAP5). It is evaluated using the subchannel analysis methodology (VIPRE-01) with low RCS flow bias conditions, including a consistent core inlet temperature corresponding to minimum RCS flow.

The NRC staff further noted that the response stated that NuScale identified a deficiency in the methodology for determining the limiting radial peaking augmentation factor. The NRC staff requested NuScale to please discuss the nature of the deficiency and asked if the deficiency affects any other events that utilize a radial peaking augmentation factor. NuScale clarified that it had made an incorrect assumption in the initial calculation. Specifically, NuScale had analyzed all possible misloaded fuel assembly configurations ("misloads") at 25 percent power to identify those that are undetectable by the in-core instrumentation. The applicant had originally assumed that the misloads that were barely within the undetectable threshold (i.e., those closest to being detected) would result in the largest radial peaking augmentation factor and therefore only calculated the radial peaking augmentation factor for those cases. However, this assumption was not correct.

NuScale acknowledged the error and revised the calculation used to investigate all undetectable misloads and identify the limiting radial peaking augmentation factor. NuScale confirmed that the error applies only to this particular event and affects no other methodologies.

RAI 9507, Question 15.04.01-5

The response states that the initial RCS flow for the event corresponds to the minimum flow rate to keep the module protection system from actuating on the low RCS flow analytical limit of 1.7 ft^3/s. However, based on an audit of the underlying calculation note, it appears the case for limiting MCHFR was run at a flow roughly two times the analytical limit. Please clarify the initial flow rate for the limiting case. NuScale responded that it intended to consider conditions down to the low RCS flow analytical limit (i.e., zero power) for the uncontrolled control rod withdrawal

from subcritical or low power analysis. However, the zero-power case isn't limiting because initial conditions, particularly initial power, drive the power overshoot. NuScale participants stated they gained a better understanding of the staff's RAI and would submit a supplemental response that includes the actual RCS flow rate used in the analysis.

RAI 9483, Question 15.01.01-7

The NRC staff found NuScale's responses to the first and third staff potential issues in this question to be satisfactory but requested clarification regarding the potential for SG overfill to degrade decay heat removal system (DHRS) capability resulting from reduced condensation heat transfer area. As stated in the original RAI, the NRC staff observed that the highest steady-state SG level may not lead to the highest transient SG level and was therefore concerned about degraded DHRS capability should the level in both SGs grow as a result of a more limiting combination of initial conditions (i.e., initial conditions that maximize the transient SG level). In response to the NRC staff's second observation in the RAI, NuScale stated that a degraded or failed single DHRS loop will not prevent the other loop from performing as required. However, the staff notes that the thermal-hydraulic analysis of the DHRS shows that DHRS performance falls rapidly between 75 and 90 percent SG level. In the overfill scenario discussed in the RAI, the level in both SGs was about 80 percent, meaning both DHRS loops could potentially be degraded.

Although the existing analysis of the overfill case showed that the DHRS was able to perform its safety function, if the transient SG level were to grow, DHRS capability could be further degraded in both trains, especially if tube plugging, fouling, and noncondensable gases were not previously considered in the analysis. For these reasons, the NRC staff requested NuScale to explain why SG overfill is not a concern for an increase in feedwater flow event, or any other event. NuScale responded that for this transient, it had calculated total heat removal through both DHRS trains. Although increasing SG level causes decreased DHRS heat removal, the total energy removed through both DHRS trains still exceeds the heat entering the SGs. Therefore, the DHRS is still removing decay heat effectively, maintaining a decrease in primary and secondary pressures. Noncondensable gases are only important at lower-pressure conditions. NuScale added that it had initiated an Engineering Change Notice (ECN) to further investigate the overfill case by assuming additional tube plugging and other biases.

RAI No. 9491

The NRC staff informed NuScale that its response to RAI No. 9491 partially resolved concerns raised in the RAI and requested clarification to resolve the following issues:

1. According to the response to RAI No. 9491, discussion of several analyses (including an analysis of a decrease in reactor coolant inventory with decreasing pressure) was deleted from FSAR Section 15.9, Rev. 2. In its RAI response, NuScale stated that the decrease in reactor coolant inventory with decreasing pressure analysis is replaced with a discussion that notes how the Module Protection System (MPS) mitigates the occurrence of divergent flow oscillations. According to the applicant's stability analysis methodology as described in Topical Report TR-0516-49417-P, "Evaluation Methodology for Stability Analysis of the NuScale Power Module" (the Stability TR), a depressurization event is analyzed to demonstrate the effectiveness of the MPS to actuate a trip and shutdown the reactor before unstable flow oscillations occur. The Stability TR provides such an analysis to illustrate that the protection of the exclusion region afforded by the MPS results in a trip and reactor shutdown prior to instability.

However, the NRC staff noted that the Stability TR analysis does not reflect the final design of the NuScale power module and associated systems. NuScale agreed to submit a revised RAI response with a revised FSAR Section 15.9 to reinclude the depressurization event consistent with the methodology described in the Stability TR.

2. Section 15.9.3 of FSAR, Revision 2, states that reactor trip is not credited in the stability operational event analysis. However, the NRC staff noted that the long term stability (LTS) solution requires operation of the MPS to enforce an exclusion region. Therefore, the applicant's LTS solution inherently credits a trip. The applicant agreed to submit a supplemental response to the RAI with revised language in the FSAR to accurately reflect the nature of the credit taken for the MPS trip. The NRC staff also requested that NuScale replace the word "credited" in section 15.9.3 statements such as, "the reactor trip is not credited in the stability operational event analysis." The NRC staff indicated that "simulated" would be an adequate and a more accurate replacement for "credited." The applicant indicated it would consider this revision as well.

U.S. NUCLEAR REGULATORY COMMISSION

SUMMARY OF JANUARY 8, 2019, JANUARY 29, 2019, AND APRIL 17, 2019

PUBLIC TELECONFERENCE WITH NUSCALE POWER, LLC

MEETING AGENDA

Tuesday, January 8, 2019

Time	Topic	Speaker
3:00 pm – 4:00 pm	Draft Final Safety Analysis Rector (FSAR) Section 19.2 Mark-up, Containment Performance Issues	NuScale/NRC

Tuesday, January 29, 2019

Time	Topic	Speaker
1:00 – 2:00 pm	Chapter 15 Requests for Additional Information (RAI) 9512, 9504, 9507 and 9483	NuScale/NRC
2:00 – 3:00 pm	Chapter 19 RAI 8840, FSAR Section 19.2	NuScale/NRC
3:00 – 4:00 pm	Chapter 15 RAI 9491	NuScale/NRC

Wednesday, April 17, 2019

Time	Topic	Speaker
3:00 – 4:00 pm	Chapter 19 RAI 8840	NuScale/NRC

LIST OF ATTENDEES

	January 8, 2019	January 29, 2019	April 17, 2019
NuScale	G. Becker B. Bristol A. Child J. Curry B. Galyean R. Gamble P. Infanger E. Mullin S. Weber	G. Becker B. Bristol S. Bristol M. Byram A. Callaway T. Codington J. Curry Y. Farawila B. Galyean R. Goff B. Hayden P. Infanger M. McCloskey E. Mullin K. Rooks D. Throckmorton S. Weber C. Williams	S. Bristol J. Curry P. Infanger E. Mullin
NRC Staff	H. Esmaili R. Franovich M. Hayes J. Martin J. Schaperow	R. Franovich R. Karas J. Schmidt A. Siwy R. Skarda	O. Ayegbusi M. Hayes O. Tabatabai
Public		S. Fields	