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

April 5, 2019

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

FROM: Getachew Tesfaye, Senior Project Manager ***/RA/***
Licensing Branch 1
Division of Licensing, Siting,
and Environmental Analysis
Office of New Reactors

SUBJECT: SUMMARY OF THE FEBRUARY 14, 2019, AND FEBRUARY 27, 2019, CATEGORY 1 PUBLIC TELECONFERENCES TO DISCUSS ACCIDENT SOURCE TERM METHODOLOGY ASSOCIATED WITH THE NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION

The U.S. Nuclear Regulatory Commission (NRC) held a Category 1 public teleconference meeting on February 14, 2019, and February 27, 2019, to discuss the NuScale Power, LLC (NuScale) accident source term methodology associated with its design certification application. These teleconferences were follow-ups to the June 7, 2018, June 27, 2018, August 9, 2018, August 29, 2018, and December 12, 2018, meetings on the same subject. The meeting attendees were personnel from NuScale and members of the general public.

The public meeting notices dated February 14, 2019, and February 27, 2019, can be found in the NRC's Agencywide Documents Access and Management Systems under Accession Nos. ML19044A499 and ML19056A434, respectively. These meeting notices were also posted on the NRC public Website.

Enclosed is the Meeting Agenda (Enclosure 1), List of Attendees (Enclosure 2), and a Meeting Overview (Enclosure 3).

Docket No.: 52-048

Enclosures:
As stated

cc w/encl.: DC NuScale Power, LLC Listserv

CONTACT: Getachew Tesfaye, NRO/DLSE
301-415-8013

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DATED: April 5, 2019

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DATE	03/06/2018	04/5/2019	04/05/2019	04/04/2019	04/02/2019	04/05 /2019

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U.S. NUCLEAR REGULATORY COMMISSION

CATEGORY 1 PUBLIC MEETING TO DISCUSS ACCIDENT SOURCE TERM METHODOLOGY ASSOCIATED WITH THE NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION

MEETING AGENDA

February 14, 2019, and
February 27, 2019

Time	Topic
1:00 p.m. – 1:15 p.m.	Introductions and Identification of Topics
1:15 p.m. – 2:20 p.m.	Discussion of Accident Source Term Methodology
2:20 p.m. – 2:30 p.m.	Public Comments/Questions
2:30 p.m.	Meeting Adjourn

U.S. NUCLEAR REGULATORY COMMISSION CATEGORY 1 PUBLIC MEETING TO DISCUSS ACCIDENT SOURCE TERM METHODOLOGY ASSOCIATED WITH THE NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION

LIST OF ATTENDEES

February 14, 2019, and
February 27, 2019

Name	Organization
Getachew Tesfaye	U.S. Nuclear Regulatory Commission (NRC)
Zachary Gran*	NRC
Ronald LaVera*	NRC
Michelle Hart	NRC
Michael Dudek	NRC
Robert Taylor	NRC
Michelle Hayes*	NRC
Anne Marie Grady	NRC
Olivia Mikula	NRC
Sheila Ray	NRC
Tania Martinez Navedo	NRC
Andy Campbell*	NRC
Jorge Cintron-Rivera*	NRC
Jason Schaperow	NRC
Edward Stutzcage	NRC
Kevin Coyne*	NRC
John Parillo*	NRC
Sam Lee**	NRC
Dan Barss	NRC
Kenny Thomas*	NRC
Mike Melton	NuScale Power, LLC (NuScale)
Gary Becker	NuScale
Peter Subaiya	NuScale
Robert Gamble	NuScale
Mark Shaver	NuScale
Scott Weber	NuScale
Paul Guinn	NuScale
Sarah Bristol	NuScale
Greg Myers	NuScale
Patrick Conley	NuScale
Bill Galyean	NuScale
Marvin Louis	Public

* Did not attend the 2-27-2019 meeting. ** Did not attend the 2-14-2019 meeting.

U.S. NUCLEAR REGULATORY COMMISSION

OVERVIEW OF THE FEBRUARY 14, 2019, AND FEBRUARY 27, 2019, PUBLIC MEETINGS

ON ACCIDENT SOURCE TERM METHODOLOGY ASSOCIATED WITH THE NUSCALE

POWER, LLC DESIGN CERTIFICATION APPLICATION

The purpose of this meeting was for the U.S. Nuclear Regulatory Commission (NRC) staff to continue the dialogue on NuScale Power, LLC's (NuScale) planned changes to the methodology for accident source term that was previously discussed in public meetings on June 7, 2018 (Agencywide Documents Management and Access System (ADAMS) Accession No. ML18173A260), June 27, 2018 (ML18206A933), August 9, 2009 (ML18240A210), August 29, 2018 (ML18249A261), and December 12, 2018 (ML18351A110).

The specific objective of the February 14, 2019, and February 27, 2019, meetings was to discuss the content of a white paper submitted by NuScale on January 31, 2019, titled "Accident Source Terms Regulatory Framework, Revision 1" (ML19032A146). NuScale gave a high-level summary of the white paper and addressed the staff's questions provided in writing as talking points prior to the meeting. The staff noted that the questions were intended to get a better understanding of the methodology and should not be construed as formal review questions of the methodology that can be used in a finding. These discussions were intended to help the staff get a high-level understanding of the NuScale's proposed approach. The staff will ultimately determine its technical viability after review and evaluation of Revision 3 of TR-0915-17565, and proposed revisions to the NuScale design certification application, when they are submitted.

NuScale informed the staff that it plans to submit Revision 3 of the TR and conforming FSAR changes for all the affected FSAR chapters, except Chapter 19, by April 17, 2019. NuScale plans to submit conforming Chapter 19 changes by May 23, 2019.

The following are the staff's questions and comments followed by NuScale's responses:

A. General (February 14, 2019):

1. The staff would like to understand more about the following statement, which is made both on pages 15 and 16 with respect to the core damage source term offsite and control room dose evaluations, respectively: "NuScale seeks agreement with NRC staff that a future licensing action could consider reduced conservatism in the CDST offsite [control room] dose evaluation."

   NuScale's Response: The DCA application will use more conservative options from the TR. However, future applicants may choose to use less conservative options from the approved TR that are not used in the current DCA. The staff noted that such an approach may unnecessarily slow down the review process and NuScale should consider including in the TR only options that are needed for the DCA application and request approval for the less conservative options in future revisions to the TR.

2. The iodine spike source term will be reviewed in more detail when the topical report is submitted, however, please provide more detail regarding the iodine spike source term. Describe at a high-level, the assumptions made in developing the source term. Does the

iodine spike source term only consider iodine or is the potential spiking of other radionuclides considered (the consideration of other radionuclides that may spike in a transient may be important for environmental qualification purposes)?

NuScale's Response: iodine concentration is estimated using RG 1.183 guidance for a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in technical specifications. The staff noted that RG 1.183 discusses iodine spiking in the context of potential offsite dose and control room dose and does not include all radionuclide contained within the building that may impact environmental qualification (EQ). The staff suggested that NuScale provide justification why other radionuclides besides iodine that may spike are not considered for EQ purposes.

3. Section 7.5 of the white paper appears to indicate that the timespan in which the under-the-bioshield radiation monitors are needed for equipment survivability space is 24 hours because this is the potential period in which core damage is progressing and conditions within the containment are rapidly changing. However, in FSAR, Table 3.11-1, regarding the environmental qualification, the monitors are required to be operational for 100 days. Please provide more information regarding why, for equipment survivability, the monitors are only needed for 24 hours and the discrepancy with the 100 days for environmental qualification. Wouldn't the monitors potentially be valuable beyond 24 hours if there was a scenario in which the event is progressing differently than expected?

   NuScale's Response: There is a fundamental difference between environmental qualification and equipment survivability. The 100 days duration is for environmental qualification based on guidance from the Institute of Electrical and Electronics Engineers (IEEE) 297 as endorsed by RG 1.47. NuScale is using the 100 days duration for environmental qualification. For equipment survivability, NuScale indicated that the guidance is very clear that the duration is based on the duration in which the equipment is needed to mitigate a severe accident. It is important to note that the 24 hours duration NuScale uses for equipment survivability, is time beyond core damage, not from the start of the transient. NuScale's research has not identified any transient that is not at a stable end stage within 24 hours after core damage.

4. In Section 7.4 regarding containment integrity, it indicates that beyond 24 hours, the containment integrity will be qualitatively assessed. If needed, is there any qualitative assessment of the under-the-bioshield monitors capability beyond 24 hours? Please explain.

   NuScale's Response: Yes, NuScale plans to provide qualitative assessment of the under-the-bioshield monitors capability beyond 24 hours. NuScale expects that the assessment will be based primarily on industry and vendor data for radiation monitors that are in current operating plants. NuScale also emphasized that they have a number of radiation monitors outside the bioshield regions that can detect radionuclide release at all possible release points.

5. Where qualitative assessment of equipment survivability is used, provide additional information regarding what the qualitative assessment will entail.

   NuScale's Response: The qualitative assessments have not yet been finalized. The intent is to use existing vendor and industry data, including doses that similar types of equipment

have been previously tested to and known and expected failure modes that the equipment has. If there is insufficient data, NuScale plans to do additional testing as the bases for survivability. All the specifics are still being finalized and will be included in the revised TR and FSAR changes. In response to the staff's question about the applicability of existing data to NuScale's design, NuScale responded that if there is doubt about the applicability of existing data, it will do its own test.

6. Any additional information that can be given to provide reasonable assurance that equipment will not unacceptably fail under the equipment survivability analysis will be helpful. For example, are synergistic effects (for example, temperature and radiation considered in the equipment survivability analysis). Are there any processes or procedures that will be used by the licensee to provide some level of assurance that there is not unexpected degradation of the penetration assemblies during operation which could challenge the survivability analysis? What provides assurance that survivability of equipment is maintained if there are any design or analytical changes?

   NuScale's Response: NuScale's plan is to evaluate temperature, pressures and radiation effects independently for equipment survivability unless there is a known synergistic effect. The staff noted that this approach differs from what was done previously by other applicants and will consider sharing that information to NuScale. Regarding procedures that will be used by the licensee to provide some level of assurance that there is not unexpected degradation of the penetration assemblies during operation, NuScale stated that licensees will follow their normal procedures to ensure containment leakage integrity. Therefore, for equipment survivability analysis, the electrical penetration assemblies (EPAs) will be in the state that is expected when they are relied on for severe accident. If additional measures are needed by future licensees, NuScale will evaluate the need for an additional COL information item as part of their survivability evaluation.

7. Please provide additional information regarding why 24 hours is used for the qualitative assessment of containment penetrations since as long as there is a higher pressure in containment than on the outside, if the penetrations were to fail it would result in a release.

   NuScale's Response: The intent of equipment survivability and containment performance goal is to prevent large early release in the first 24 hours and unmitigated release after that point. Therefore, the goal of survivability evaluation is not to prevent release in the later stages of an accident. Based on vendor information, the EPAs are not expected to experience catastrophic failure within 24 hours. Additionally, NuScale's research on severe accidents has shown that within 24 hours after core melt, the radionuclides that have been released would have been substantially deposited on the containment surfaces. Therefore, if there is degradation of the EPAs after 24 hours, it will not result in large release.

B. Environmental Qualification (EQ) Specific (February 14, 2019):

1. 10 CFR 50.49 states, in part:

   The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.

   10 CFR 50.49 requires the applicant to establish a program for qualifying the electric equipment important to safety located in harsh environment. IEEE Std 323-1974 as endorsed by RG 1.89 describes the methodology accepted by the staff to qualify electrical

equipment important to safety located in a harsh environment. Environmental qualification is an assessment of equipment functionality during and after DBA. Whereas, equipment survivability evaluates equipment functionality during and after a severe accident. In other words, environmental qualification is a well-defined, established process that could be more rigorous than equipment survivability.

Based on the above, please describe the methodology used to assess equipment survivability for the NuScale design. Will equipment survivability be determined by testing, analysis, or testing in combination with analysis. If only analysis is used, please discuss the technical basis for the specific equipment. Discuss for EPAs and post-accident monitoring (PAM).

NuScale's Response: We are still developing the equipment survivability evaluation exact approach for different components. We plan to adopt approaches previously used such as in the CE System 80+ survivability evaluation. In response to the staff's question on PRA insight for equipment survivability, NuScale stated that there is currently a discussion in FSAR Section 19.2 about extreme temperatures and pressures for equipment survivability but not for extreme radiological conditions since that was not an issue prior to the planned revision to the accident source term. The planned revision to FSAR Section 19.2 will address PRA insight on equipment survivability for the revised accident radiological source term that is not covered by FSAR Section 3.11 environmental qualification.

2. 10 CFR 50.49 requires the equipment to perform under conditions existing during and following design basis accidents. The applicant stated that the survivability evaluation will be performed to demonstrate equipment will survive for 24 hours maximum duration, and analysis thereafter. Additional basis for the 24 hours maximum duration is needed. How will survivability be demonstrated for the specific equipment (i.e., EPA, PAM, and containment isolation valves (CIVs)) and how will the survivability evaluation show equipment function, if the duration of the equipment is greater than 24 hours?

   NuScale's Response: This is addressed in Part A above.

3. Please provide a list of equipment that will be assessed for equipment survivability. Will there be changes to the EQ list in NuScale FSAR Section 3.11? If the answer is yes, please discuss the changes.

   NuScale's Response: The list of items that will be assessed for equipment survivability include equipment that support functions that are needed for severe accident mitigation such as containment isolation valves, EPAs, post-accident monitoring, and equipment used to monitor combustible gases. There is no plan to change the EQ list in FSAR Section 3.11.

C. Additional questions discussed on February 27, 2019.

1. The staff needs to have a discussion with NuScale to clarify their proposed approach on 10 CFR 50.34(f)(2)(vii) compliance. In particular, the staff asks NuScale to clarify whether they intend the entirety of 10 CFR 50.34(f)(2)(vii) to be assessed using the core melt source term or whether part of 10 CFR 50.34(f)(2)(vii) is being met using the iodine spike source term. The white paper indicates that control room dose is being addressed with the core melt source term, but it isn't clear regarding other aspects of 10 CFR 50.34(f)(2)(vii), particularly as it relates to equipment protection.

In addition, FSAR Chapter 12, Section 12.2.1.13, Revision 2, indicates that 10 CFR 50.34(f)(2)(vii) is being evaluated to the iodine spike source term. Specifically, FSAR Section 12.2.1.13 references 10 CFR 50.34(f)(2)(vii) and discusses accident source terms and discusses several accident source term tables. NuScale indicated previously in a clarification call this past December that this FSAR Chapter 12 information has been revised to reflect the iodine Spike source term. Therefore, it would appear that the design basis iodine spike source term is being relied on for at least part of 10 CFR 50.34(f)(2)(vii) compliance.

NuScale's Response: We realize FSAR Revision 2 is inconsistent with the proposed path forward, and it will be updated in the next revision. NuScale's approach to comply with 10 CFR 50.34(f)(2)(vii) is through use of core damage source term. As noted by the staff, NuScale discussed the implication of that in the whitepaper. To meet the regulation, the control room dose is being addressed with the core melt source term. The FSAR will be revised to reflect the exemption request submitted for post-accident sampling. The part of 10 CFR 50.34(f)(2)(vii) requirement for protecting safety equipment from the radiation environment is being addressed with the core melt source term using equipment survivability evaluation.

2. TR-0716-50424-P, Revision 0, "Combustible Gas Control," and NuScale's response to RAI 8862, Question 6.2.5-2, states that the following components are required to mitigate severe accident conditions.

TR Table 3-16

<table>
<tr><th>Component</th><th>Performance Requirement</th><th>Reflected Detonation Treatment</th><th>DDT Load Treatment</th></tr>
<tr><td>CIVs</td><td>Remain closed</td><td rowspan="5">ASME Service Level C</td><td rowspan="5">ASME Service Level D</td></tr>
<tr><td>ECCS main valves</td><td>Remain open</td></tr>
<tr><td>ECCS trip and reset valve</td><td rowspan="2">Preserve containment structural integrity</td></tr>
<tr><td>Electrical penetration</td></tr>
<tr><td>Cables and wiring</td><td>Do not generate debris</td><td colspan="2">Environmental Qualification</td></tr>
</table>

Please confirm that the above table is accurate and complete.

NuScale's Response: We have not completed identifying the components that should be on the equipment survivability list to mitigate severe accidents as part of the FSAR Chapter 19 review. The components listed in the combustible gas control TR table is to comply with 10 CFR 50.44(c) for inside containment. The list that will be generated as part of the accident source term methodology TR revision may not have a one-to-one correspondence with the list for combustible gas control.

D. Next steps:

1. NuScale submits Revision 3 of the accident source term methodology TR and conforming FSAR changes for all the affected FSAR chapters, except Chapter 19, by April 17, 2019.

2. NuScale submits conforming Chapter 19 changes by May 23, 2019.