UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

December 20, 2019

Ms. Margaret M. Doane
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SAFETY EVALUATION OF THE NUSCALE POWER, LLC TOPICAL REPORT TR- 0915-17565, REVISION 3, "ACCIDENT SOURCE TERM METHODOLOGY," AND SOURCE TERM AREA OF FOCUS REVIEW FOR THE NUSCALE SMALL MODULAR REACTOR

Dear Ms. Doane:

During the 669th meeting of the Advisory Committee on Reactor Safeguards, December 4-6, 2019, we completed our review of the NRC staff's safety evaluation report (SER) of NuScale Power, LLC (NuScale) topical report TR-0915-17565, Revision 3, "Accident Source Term Methodology." We also completed our source term area of focus review for the NuScale design certification application (DCA) as discussed in our letter to you on September 25, 2019. Our NuScale Subcommittee also reviewed these matters on November 20, 2019. During these meetings, we had the benefit of discussions with NuScale and the staff. We also had the benefit of the referenced documents.

This letter completes our focus area review of source term for the NuScale DCA. A finding relative to the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 52.53 awaits completion of all source term-related chapters as well as the remaining focus area reviews.

CONCLUSIONS AND RECOMMENDATIONS

1. Given its unique design attributes, the NuScale DCA uses alternative source terms for both normal operation and accident conditions for siting, safety analysis, control room and technical support center habitability, and equipment qualification and survivability. This approach aligns with Commission guidance noting that design-specific source terms for light-water small modular reactors may not necessarily follow all guides that currently pertain to large light-water reactors (LWRs).

2. The exclusion area boundary (EAB) and low-population zone (LPZ) are anticipated to be close to a NuScale plant. The traditional dose model to calculate radiological consequences for LWRs is inaccurate at short distances from the reactor. Therefore, NuScale has

modified an NRC computer code that is more accurate at these reduced distances to address dose evaluations at the EAB and LPZ. The staff has found this approach acceptable.

3. The overall approach to establish the source term for NuScale is acceptable with the conditions and limitations noted by the staff. The SER on the topical report methodology should be issued.

4. Important design differences in NuScale compared to a conventional pressurized-water reactor call into question the prescriptive application of the post-accident requirements for long-term hydrogen and oxygen monitoring. The risk tradeoff between unisolating the NuScale containment to enable long-term hydrogen and oxygen monitoring should be weighed against alternatives that may not require such monitoring. We will continue to explore this issue in our NuScale review.

BACKGROUND

The impact of radiological source terms during normal operation and accidents must be assessed in safety analyses for reactor designs. Because of some of its unique design attributes, NuScale has selected alternative source terms for both these conditions. NuScale in their topical report TR-0915-17565, Revision 3, "Accident Source Term Methodology," describes a general methodology for developing accident source terms for their design. These normal and accident condition source terms are used for siting and safety analysis, control room habitability, technical support center habitability, and equipment qualification and survivability.

DISCUSSION

Source Term for Normal Operation

NuScale uses a failed fuel fraction of 66 parts-per-million (ppm) to establish reactor effluent source terms under normal operation. This corresponds to less than one failed fuel rod in a NuScale Power Module (NPM) core. This value is based on operational experience of the existing LWR fleet.

Source Term for Design-Basis Events

NuScale uses 10 times the failed fuel fraction value (660 ppm) to establish the design basis primary coolant activity that is used in shielding analysis and for design-basis accidents (DBAs). This level of activity corresponds to the failure of about 6-7 fuel rods in the NPM core. NuScale includes an iodine spike (washout of iodine deposited on the inside of cladding) within this source term that is meant to bound expected releases during a DBA. It is an empirical factor of 500 increase in iodine release rate for eight hours. The resultant source term is used to evaluate DBAs including a control rod ejection accident, a main steam line break outside of containment, a steam generator tube rupture, and small breaks outside of containment.

This approach is consistent with the intent of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors." The highest doses predicted for the design-basis events are those associated with the fuel handling accident. The staff has found this approach to be conservative and acceptable.

NuScale proposed that this DBA source term is also appropriate for use in evaluating equipment qualification. Although inconsistent with language in Regulatory Guides 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” and 1.183 stating that a core damage source term should be used in evaluating equipment qualification, NuScale asserted that none of the DBAs for the NPM result in core damage. The staff agreed with this assertion, observing that it is consistent with language in regulation (10 CFR 50.49(e)(iv)) and General Design Criteria 4, “Environmental and Dynamic Effects Design Bases.”

As part of our discussions, we observed that there is the potential for small particulates due to cladding fragmentation and dispersion to be released during postulated small break loss-of-coolant accident (LOCA) events. The level sensor in the containment is used for initiating the emergency core cooling system. Therefore, the presence of such particulates should be evaluated as part of the equipment qualification Inspections, Tests, Analyses, and Acceptance Criteria for its acceptability.

Core Damage Source Term

Historically, large source terms were postulated for siting using the concept of maximum hypothetical accident. For LWRs, this maximum hypothetical accident is assumed to be a large break LOCA with substantial core melt. The associated dose source term is described in TID-14844, “Calculation of Distance Factors for Power and Test Reactors.” After much research following the accident at Three Mile Island, an improved understanding of radionuclide behavior in severe accidents led to a revised accident source term for LWRs with its technical basis described in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants.”

The NuScale plant does not have large reactor coolant system piping so a large break LOCA as the bounding accident scenario would not appropriately reflect the safety response of their plant to a severe event. Furthermore, as stated in SECY-16-0012, “Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors,” small modular reactor and non-light water reactor applicants can employ modern analysis tools to demonstrate quantitatively the safety features of those designs. Modern source term analysis methods can also be used by applicants to demonstrate the ability of the enhanced safety features of plant designs to mitigate accident releases. Design-specific accident source terms for light-water small modular reactors may not necessarily follow all guides that currently pertain to large LWRs.

Thus, NuScale uses a core damage source term based on several severe accident scenarios that are selected to encompass most of the risk dominant sequences for their design. The severe accident progression and associated source term were calculated by MELCOR to derive a surrogate radiological source term into the containment for a core damage event. Fission product transport and removal in the containment is calculated using MELCOR thermal hydraulics and only natural phenomena (agglomeration/coagulation, settling, thermophoresis and diffusiophoresis) in the STARNAUA code.

To assess iodine re-evolution inside containment, a pH post-accident calculation is performed by NuScale considering all relevant acids and bases that could be generated in the containment water. Little iodine evolution is expected at the pH of the pool in the postulated core damage events.

This core damage source term is used for siting evaluations (EAB and LPZ) in Chapter 2, "Site Characteristics and Site Parameters," and for equipment survivability assessment in Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation," of the DCA. The staff performed a detailed review of the methodology and completed independent confirmatory analyses using MELCOR. The staff found the NuScale approach acceptable for developing the core damage source term.

Radiological Dose Consequence Calculations

The EAB and LPZ are anticipated to be close to a NuScale plant. The traditional PAVAN model to calculate radiological consequences from LWRs is inaccurate at short distances from the reactor. NuScale has modified the ARCON96 code that is more accurate at these reduced distances to address dose evaluations at the EAB and LPZ. The staff found their modifications and approach acceptable.

Source Term Focus Area Review

As part of our focus area review, the staff presented their resolution to source term related open items in the relevant chapters of the SER. These items are related to regulations for post-accident sampling and hydrogen and oxygen monitoring. Staff discussed the technical basis for NuScale's requested exemption from post-accident sampling of the reactor coolant and containment atmosphere. The staff also discussed their evaluation of the possibility of leakage from the post-accident hydrogen and oxygen monitoring system and their dose evaluations performed to assess potential consequences if this were to occur following the postulated core damage event.

For post-accident sampling, the staff evaluated the response of the NuScale design to the core damage event and determined that each of the sampling functions important for establishing the condition of the core can be provided by other means. Thus, the staff evaluated the requested exemption from post-accident manual sampling for the NuScale design. The staff's evaluation of this exemption request can be found in Section 9.3.2, "Process Sampling System," of the SER.

NuScale has not provided complete design information of the following items as part of the DCA:

- Radiation shielding for the NPM to Reactor Building Steam Gallery penetrations;

- Design features and analysis for leakage of systems outside of containment (containment evacuation system, containment flood and drain system, and post-accident sampling system) used for combustible gas monitoring, that may have highly radioactive fluid following an accident.

The staff proposes that these items be addressed by a future combined license (COL) applicant.

We have concerns related to the need and capability for long-term post-accident hydrogen and oxygen (combustible gas) monitoring. The NuScale design has incorporated hydrogen and oxygen monitoring connected from the containment evacuation system piping. We cannot conclude based upon the available design information that the monitoring system will provide accurate data on the actual hydrogen and oxygen concentration in containment. Most of the system downstream of the isolation valves is neither safety grade nor seismically qualified.

Opening valves to monitor the containment atmosphere for hydrogen and oxygen would spread contamination to these non-safety grade components. The staff has estimated that radiological releases associated with leakage from the monitoring system could be significant. Therefore, we are concerned that opening the isolation valves may spread contamination and introduce radiological dose consequences. We will address these issues in more detail when we evaluate the design of this system as part of our Phase 5 NuScale Chapter 6, "Engineered Safety Features," and 9, "Auxiliary Systems," SER reviews.

We also question if actual safety benefits accrue from direct application of the post-accident hydrogen and oxygen monitoring requirements to the NuScale design. These requirements were developed to assess challenges to containment integrity and hydrogen deflagration/detonation risk following the onset of a severe accident given the large volume of air in pressurized-water reactor containments. The rationale for these requirements does not apply for the NuScale design because:

a) the containment vessel has a high design pressure (1050 psia) and initially contains no air;

b) it takes weeks before combustible levels of hydrogen and oxygen can be generated via radiolysis giving the operators significant time for countermeasures well before any monitoring information would be available to inform potential mitigating actions;

c) the design allows for inerting the containment environment with nitrogen through the chemical and volume control system; and

d) there are other pressure, temperature and radiation sensors that may be used to follow severe accident progression without the hydrogen and oxygen monitoring system.

Thus, the important design differences in NuScale compared to a conventional pressurized-water reactor call into question the prescriptive application of the requirements for long-term hydrogen and oxygen monitoring in the NuScale design. The risk tradeoff between unisolating the NuScale containment to enable long-term hydrogen and oxygen monitoring should be weighed against alternatives that may not require such monitoring for these low risk events (e.g. inerting the containment with nitrogen soon after the severe accident occurs based on temperature and pressure instrumentation in the containment and radiation instrumentation under the bioshield). This type of comparative evaluation will be consistent with the guidance in the Staff Requirements Memorandum for SECY-19-0036, "…the staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria … is unnecessary to provide reasonable assurance of adequate protection of public health and safety."

Sincerely,

/RA/

Peter Riccardella,
Chairman

REFERENCES

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3. U. S. Nuclear Regulatory Commission, SECY-19-0079, “Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application,” August 16, 2019 (ML19107A455).

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8. NuScale Power, LLC, “NuScale Power, LLC Response to NRC Request for Additional Information No. 9690 (eRAI No. 9690) on the NuSCale Topical Report, ‘Accident Source Term Methodology,’ TR-0915-17565, Revision 3,” September 5, 2019 (ML19248D680).

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10. Advisory Committee on Reactor Safeguards, “Proposed Focus Area Review Approach of the Advanced Safety Evaluation Report With no Open Items for the Design Certification Application of the NuScale Small Modular Reactor,” September 25, 2019 (ML19269B682).

11. U. S. Nuclear Regulatory Commission, “Review Schedule for the NuScale Power, LLC, Standard Design Certification of a Small Modular Reactor,” May 22, 2017 (ML17103A380).

12. U. S. Nuclear Regulatory Commission, Staff Requirements Memorandum for SECY-19-0036, “Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” July 2, 2019 (ML19183A408).

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