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

June 19, 2019

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

SUBJECT: INTERIM LETTER – CHAPTER 3, SECTION 3.9.2, AND CHAPTERS 14, 19 AND 21 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE DESIGN CERTIFICATION APPLICATION REVIEW OF THE NUSCALE SMALL MODULAR REACTOR

Dear Ms. Doane:

During the 664th meeting of the Advisory Committee on Reactor Safeguards, June 5-7, 2019, we met with representatives of NuScale Power, LLC (NuScale) and the NRC staff to review Chapter 3, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures and Components;" Chapter 14, "Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria;" Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors;" and Chapter 21, "Multi-Module Design Considerations," of the safety evaluation report (SER) with open items associated with the NuScale design certification application (DCA). Our NuScale Subcommittee also reviewed these chapters on May 14-16, 2019. During these meetings, we had the benefit of discussions with NuScale and the NRC staff. We also had the benefit of the referenced documents. Note that Chapter 19, Section 19.4, "Strategies and Guidance to Address Loss of Large Areas of the Plant Because of Explosions and Fire," of the DCA is evaluated as Section 20.2 by the staff and will be reviewed as part of Chapter 20 at a later date.

CONCLUSIONS AND RECOMMENDATIONS

1. The TF-3 comprehensive vibration tests are required to ensure that the steam generator design is not susceptible to flow-induced vibration. The completion of these tests should be identified as an item for Inspections, Tests, Analyses and Acceptance Criteria (ITAAC).

2. We have not identified any major issues at this time for Chapter 3, Section 3.9.2, and Chapters 14, 19 and 21.

3. To help identify risk insights in this unique design, there are technical issues in the probabilistic risk assessment (PRA) that merit further consideration.

BACKGROUND

NuScale submitted a DCA for its small modular reactor on December 31, 2016. The staff's Phase 2 SER chapters related to the DCA include open items. In addition to a description of the staff review and its bases for acceptance of the DCA, the SER chapters also identify the information a combined license applicant must provide.

Our review is being conducted on a chapter-by-chapter basis to identify technical issues that may merit further consideration by the staff. This process can aid in the resolution of concerns and facilitates timely completion of the design certification application review. Our review addresses the staff's SER and DCA Chapter 3, Section 3.9.2, Revision 2; Chapter 14, Revision 2; Chapter 19, Revision 2; and Chapter 21, Revision 1, as well as supplementary material, including responses to staff requests for additional information.

DISCUSSION

For this interim letter, we make the following observations on selected elements of the design addressed in these chapters.

DCA Chapter 3, Section 3.9.2 – Dynamic Testing and Analysis of Systems, Components and Equipment

This section of the Chapter 3 SER reviews the analytical methodologies, testing procedures, and dynamic analyses that the applicant used to ensure the structural and functional integrity of the piping systems, mechanical equipment, reactor vessel internals (RVIs), and their supports under dynamic loadings, including those caused by fluid flow and postulated seismic events.

In its review, the staff focused on dynamic system analysis of the reactor internals under service level D conditions and the applicant's reactor internals comprehensive vibration assessment program. In addition to Section 3.9.2, the staff review covered Appendix 3A of the DCA and four associated NuScale technical reports.

The staff identified several issues and open items in its review of NuScale's service level D dynamic system analyses. These concerned assumptions regarding system damping; fluid gap between the core barrel and reflector blocks; acoustic absorption coefficient for the reactor pool floor; generation of in-structure seismic response spectra; adequacy of NuScale power module (NPM) seismic analysis cases; uplift of reflector blocks; and seismic analysis details of major RVI components, including the steam generators. These items are either closed or in the process of being resolved.

Regarding NuScale's RVI vibration assessment program, the staff performed detailed evaluations of components considered most susceptible to flow-induced vibration (FIV) concerns, including the helical coil steam generator tubes and supports, the steam generator tube inlet flow restrictors, the control rod drive shafts, the in-core instrument guide tubes, and the NPM primary and secondary coolant piping.

Although the natural circulation NPMs have significantly lower primary coolant flow rates than conventional PWRs, some RVI components contain long rods and tubes that may be susceptible to FIV. Acoustic resonance may also be possible in dead piping legs adjacent to secondary coolant flow. The staff review identified some non-conservatisms in the NuScale FIV analyses that may outweigh the conservatisms. These are being addressed by testing. The testing focuses on key FIV mechanisms with low margins of safety and high uncertainty, plus initial startup testing with online vibration monitoring performed in accordance with ASME standards. Prior to and following initial startup testing, components will be inspected for mechanical wear and signs of vibration-induced damage.

To date, NuScale has completed two sets of preliminary vibration tests (TF-1 and TF-2) for the steam generators. A third set of integral tests (TF-3) is planned to obtain additional data. The TF-3 tests are crucial to provide the basis for reasonable assurance against susceptibility to FIV, and they may not be completed before scheduled issuance of the design certification. The staff is therefore conducting a detailed review and onsite audit of plans for this test program. The successful completion of the TF-3 tests should be identified as an ITAAC.

DCA Chapter 14 – Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria

The initial test program consists of a series of preoperational and startup tests conducted by the startup organization. Preoperational testing is conducted for each NPM following completion of construction testing but prior to fuel load. Completion of preoperational testing for each NPM is necessary to ensure the NPM is ready for fuel loading and startup testing. Additional tests of each NPM are performed following the completion of preoperational testing. Startup testing includes initial fuel loading and pre-critical testing, initial criticality testing, low-power testing and power-ascension testing.

The scope of the ITAACs addressing these tests needs to be sufficient to provide reasonable assurance that, if the ITAACs are successfully completed, the facility has been constructed and can be operated in accordance with the regulations, and the combined license. We concur with the staff that, pending completion of the confirmatory items and closure of the open items, NuScale has demonstrated compliance with the NRC regulations regarding its initial test program. In addition, we concur with the staff that the open items preclude finalization of conclusions related to ITAACs.

DCA Chapter 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

Section 19.1

This section describes the PRA performed for the NuScale design and summarizes the Level 1 and Level 2 PRA, which evaluates the risk associated with all modes of operation for both internal and external initiating events. Major topics include: PRA quality, design features to minimize risk, methodology, data, uncertainties, sensitivities, insights, and results. Internal and external event PRA for at-power and other modes of operation is described, and the risk associated with multiple modules is also discussed[1]. A seismic margins analysis was performed rather than a seismic PRA. At this stage, the PRA scope is complete and sufficient for the consideration of risk results.

The PRA was integral to the design process, and risk insights influenced a number of design decisions. This integrated process contributed to achieving low values of core damage frequency (CDF) and large release frequency (LRF). These low values provide confidence that the NuScale design meets the Commission's Safety Goals with margin.

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[1] The PRA was performed for a single module. The likelihood to fail more than one module was approximated.

In fact, the NuScale PRA results indicate that the risk associated with the plant operation is apparently negligible. To further build confidence that these results accurately reflect the risk, we have identified the following technical issues that may merit further consideration by the staff.

Emergency Core Cooling System (ECCS) Valves Model and Data

Failures of NuScale's unique ECCS valves are among the most important risk contributors identified in the PRA. These valves are closed by hydraulic pressure and opened by spring actuation. The PRA failure model for these valves is not available in Chapter 19, but insights from some of the significant CDF cutsets suggest a possible non-conservative modeling of the valves passive opening at low differential pressure. We plan to visit NuScale for a further examination of the valve design and the associated PRA model to help build confidence that the plant risk is accurately represented.

Uncertainty Analysis

The applicant reported very low numerical values for the CDF and LRF. These values are based on available information but should have high uncertainty due to: (1) incomplete design and construction, undeveloped procedures, and a lack of operating experience as expected for any new design; and (2) lack of data on reliability of the design-unique and risk-significant components (e.g., the ECCS valves), or reliability of passive heat transfer to the reactor pool.

However, the results reported in Section 19.1 are not consistent with the expectation of high uncertainty. Even though the presented uncertainty results do not account for the model uncertainties, but only include parameter uncertainties, that alone would not explain these narrow uncertainty ranges. To ensure that the risk measures reported in the PRA include more realistic uncertainty results (and the associated mean values), the uncertainty analysis merits further investigation.

Sensitivity Analyses

Sensitivity analyses were performed to provide additional insights on the risk results and component importance measures, and to investigate the importance of modeling assumptions and uncertainties. As with uncertainty, these evaluations are especially important given the reported low numerical value for the risk measures. We found these sensitivity analyses incomplete in the following areas:

- No combinations of sensitivity results for different assumptions are included. The PRA results could be especially sensitive to a combination of specific assumptions and uncertainties in the same accident sequence.

- Some sensitivity cases are general and not event-specific; thus, their results may obscure relevant risk insights. For example, the sensitivity analysis on common cause failures (CCFs) selects a failure rate of 0.002 for all events and concludes that Safety Goals are still met. Nevertheless, the risk is not negligible in this case, and risk insights are different. A realistic sensitivity analysis could provide more meaningful insights; e.g., CCF for ECCS and decay heat removal system valves or degradation of passive cooling systems.

Based on the above discussion, more complete sensitivity analyses would increase confidence that the plant risk is accurately represented.

Human Errors of Commission

In the SER, the staff stated that the reactor building crane (RBC) analysis considered human errors of commission that may cause an initiating event leading to crane failure. The staff review revealed that this is the dominant failure causing an initiating event. Even though we agree with the reason for excluding detailed hardware failures from the RBC analysis (i.e., the lack of final design details), we believe that this operator error should be identified and included as a risk-significant human action in Table 19.1-70, and also included in Chapter 18.

Section 19.2

This section describes and analyzes the prevention and the mitigation of severe accidents. The section discusses severe accident prevention and the design's capability to prevent specific severe accidents, including those resulting from beyond-design-basis events such as an anticipated-transient-without-scram event, fire protection issues, station blackout, and an interfacing system loss-of-coolant accident.

NuScale evaluated a module's response to the aforementioned spectrum of beyond-design-basis events. Results emphasize the capability of the NuScale design to mitigate severe accidents and phenomena, such as hydrogen generation, high-pressure melt ejection, in-vessel steam explosions, induced steam generator tube failure, and equipment survivability.

NuScale performed severe accident simulations using a modified version of MELCOR to incorporate the NPM unique design features. For severe accidents without containment bypass, results indicate that relocated core materials would be retained within the reactor vessel. NuScale reviewed cases with significant core relocation and selected parameters that they believed would bound heat transfer from the relocated core materials to the reactor vessel. The NuScale evaluation finds that containment integrity would not be challenged in a severe accident.

The staff completed audit calculations using an independently developed input model with a version of the MELCOR code that had been similarly modified. The staff selected three representative severe accident sequences for analysis comparison. For each scenario, no significant differences were found in comparison with NuScale's analysis of severe accident mitigation. The staff also concluded that NuScale had considered an appropriate range of credible core damage scenarios.

Although the MELCOR results provided confidence in the module's response during beyond-design-basis events, the staff observed that the success of in-vessel retention could be impacted by phenomenological uncertainties, such as the potential for stratification of fuels and metals within relocated core materials. Nevertheless, the staff concluded that the potential for large radiation releases from such events would be mitigated by the containment vessel. Were the containment lower head to fail, releases from relocated core debris would be scrubbed sufficiently by the deep reactor pool water to preclude a large release.

The staff has noted that the original design for the bio-shield could allow for the accumulation of hydrogen under the bio-shield during a core-damage event to flammable or combustible concentrations. Such hydrogen combustion may also affect other modules during severe accidents. Consequently, NuScale initiated a bio-shield redesign project to alleviate this concern. The redesign is currently under review by the staff, and it is an open item.

The staff also continues to interact with NuScale regarding the qualification of electrical penetration assemblies for radiation doses associated with core-damage accident scenarios. NuScale recently identified an issue with its evaluation of these assemblies, and the staff is tracking this as an open item as part of its review of the accident source term methodology topical report under revision.

DCA Chapter 21 – Multi-Module Design Considerations

This chapter seeks to demonstrate that the safety-related systems and functions that prevent or mitigate NPM design-basis accidents are not adversely affected as a result of failures of shared (common) systems among NPMs. The applicant discusses the design measures taken to ensure those systems do not introduce significant multi-module risks. The applicant concludes that an accident in one NPM does not result in an accident in another NPM and that no design-basis accidents result from operation or failure of shared systems. The staff has reviewed the information presented by the applicant and documented its findings and conclusions as part of the Chapter 15 review.

NuScale provides information regarding the number of modules supported by various shared systems. However, the documentation is unclear whether all the supported modules must be shut down when a shared system becomes unavailable. We will continue exploring this topic in upcoming meetings with the staff and NuScale.

Summary

The TF-3 comprehensive vibration tests are required to ensure that the steam generator design is not susceptible to flow-induced vibration. The completion of these tests should be identified as an ITAAC. We have not identified any major issues at this time for Chapter 3, Section 3.9.2 and Chapters 14, 19 and 21. To help identify risk insights in this unique design, there are technical issues in the PRA that merit further consideration.

Members Corradini and Rempe did not participate in Chapter 19, Section 19.2 deliberations.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

REFERENCES

1. U.S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation with Open Items for Chapter 3, Section 3.9.2, ‘Dynamic Testing and Analysis of Systems, Components, and Equipment’,” April 17, 2019 (ML19102A109).

2. NuScale Power, LLC, “Design Certification Application – Chapter 3, ‘Design of Structures, Systems, Components and Equipment, Sections 3.9 to 3.13’,” Revision 2, October 30, 2018 (ML18310A323).

3. NuScale Power, LLC, “Design Certification Application – Chapter 3, ‘Design of Structures, Systems, Components and Equipment, Appendices 3A to 3C’,” Revision 2, October 30, 2018 (ML18310A324).

4. U.S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation with Open Items for Chapter 14, Section 14.2, ‘Initial Plant Test Program – Design Certification and New License Applicants’,” April 15, 2019 (ML19092A423).

5. U.S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation with Open Items for Chapter 14, Section 14.3, ‘Inspections, Tests, Analyses, and Acceptance Criteria’,” May 21, 2019 (ML19066A189).

6. NuScale Power, LLC, “Design Certification Application – Chapter 14, ‘Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria’,” Revision 2, October 30, 2018 (ML18310A336).

7. U.S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation with Open Items for Chapter 19, ‘Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors’,” April 19, 2019 (ML19073A071).

8. NuScale Power, LLC, “Design Certification Application – Chapter 19, ‘Probabilistic Risk Assessment and Severe Accident Evaluation’,” Revision 2, October 30, 2018 (ML18310A342).

9. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With Open Items for Chapter 21, “Multi-Module Design Considerations,” April 19, 2019 (ML18220B313).

10. NuScale Power, Design Certification Application, Chapter 21, “Multi-Module Design Considerations,” Revision 2, October 30, 2018 (ML18310A344).

11. NuScale Power, LLC, “Technical Report, ‘NuScale Power Module Seismic Analysis’,” TR-0916-51502, Revision 2, April 2019 (ML19093B850 Publicly Available/ML19094B831 Non-Publicly Available).

12. NuScale Power, LLC, “Technical Report, ‘NuScale Power Module Short-Term Transient Analysis’,” TR-1016-51669, Revision 0, December 2016 (ML17005A132).

13. NuScale Power, LLC, “Technical Report, ‘NuScale Comprehensive Vibration Assessment Program’,” TR-0716-50439, Revision 1, January 2018 (ML18022A221 Publicly Available/ML18022A220 Non-Publicly Available).

14. NuScale Power, LLC, “Technical Report, ‘NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan’,” TR-0918-60894, Revision 0, December 2018 (ML18341A337 Publicly Available/ML18344A069 Non-Publicly Available).

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