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

May 30, 2019

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

SUBJECT: INTERIM LETTER: CHAPTERS 4 AND 5 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 663rd meeting of the Advisory Committee on Reactor Safeguards, May 2-3, 2019, we met with representatives of NuScale Power, LLC (NuScale) and the NRC staff to review Chapter 4, "Reactor," Chapter 5, "Reactor Coolant System and Connecting Systems," and the associated technical topical report, TR-1015-18177, "Pressure and Temperature Limits Methodology" (PTLM) 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 April 17, 2019. During these meetings, we had the benefit of discussions with NuScale and the staff. We also had the benefit of the referenced documents.

CONCLUSIONS and RECOMMENDATIONS

1. There are a number of Chapter 4 and Chapter 5 open items related to Chapter 15 accident analysis issues that must be reviewed and resolved in order to demonstrate acceptability of the NuScale reactor design in satisfying General Design Criteria (GDC) 27, 34 and 35.

2. We have not identified any additional major issues at this time for Chapter 4 and Chapter 5.

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 their 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. Accordingly, the staff has provided Chapters 4 and 5 of the SER with open items for our review. The staff's SER and our review of these chapters addresses DCA Chapter 4, Revision 1, Chapter 5, Revision 2, and the associated topical report PTLM report, Revision 0, as well as supplementary material including responses to staff requests for additional information (RAIs).

DISCUSSION

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

DCA Chapter 4 – Reactor

This chapter describes the reactor and the reactor core design, the fuel rod and fuel assembly design, the core control and monitoring components, and the nuclear and thermal-hydraulic design. The fuel rod and fuel assembly design features, analyses and anticipated performance have been adapted from PWR fuel technology currently in-service in the operating fleet. The operational linear power levels are below current designs and fuel assembly limits are within the operating fleet experience. Modern design features have been adopted to address fuel failure mechanisms.

NuScale applied approved reactor analysis methods for predicting reactor core and fuel performance. We previously concurred with staff conclusions that the applicant’s nuclear analysis methods, critical heat flux correlations, and subchannel analysis methodology are acceptable. The use of commercial fuel under NuScale operating conditions offers larger margins than current PWRs. However, there are a number of open items discussed in the SER related to the NuScale reactor design response to Anticipated Operational Occurrences, (AOOs) and accidents and whether the reactor design meets the requirements of GDC 27 “Combined Reactivity Control Systems Capability” and GDC 35 “Emergency Core Cooling.”

In Chapter 4, NuScale has chosen to establish the shutdown margin based on normal operation conditions. However, following reactor shutdown, operators can control the cooldown of the reactor or, for some scenarios, may rely on passive, Decay Heat Removal System (DHRS) cooling. DHRS operation results in lower moderator temperature than manual operation using standard procedures. Lower moderator temperatures may result in lower shutdown margins. It would be prudent to evaluate long-term reactivity at the lower moderator temperatures resulting from DHRS operations as opposed to normal shutdown temperatures.

The staff is evaluating NuScale responses to RAIs on the ability of the reactor design to maintain long-term reactivity control following AOOs or postulated accidents. We agree that these Chapter 15 accident analysis issues need to be reviewed and resolved to demonstrate acceptability of the NuScale reactor design in meeting GDC 27 and GDC 35.

NuScale defines Operating Mode 4 as that mode required prior to transport of the power module to the refueling station, and this transport operation may be an important contributor to risk. It would be prudent to provide additional margin to criticality by specifying, in the core operating limits report, that the refueling-mode boron concentration be established before the reactor state is changed from Mode 3 (safe shutdown) to Mode 4.

<u>DCA Chapter 5 – Reactor Coolant System and Connecting Systems</u>

This chapter describes the reactor coolant system (RCS) design, which provides for the circulation of the primary coolant. The reactor design relies on natural circulation flow of the water coolant and does not include reactor coolant pumps or an external piping system. The RCS is a subsystem of the NuScale Power Module and includes the reactor vessel, the integral pressurizer, the reactor vessel internals, the reactor safety valves, two steam generators integrated into the reactor vessel, the DHRS, and RCS piping inside the containment and associated RCS instrumentation.

The NuScale helical coil steam generator design, with secondary-side steam generation inside the tubes and primary system pressure external to the tubes, is unique. The steam generator is integral to the upper reactor vessel structure. The selection of a thermally treated alloy 690, (690TT), for the tubing material is appropriate based on performance in current PWRs. While alloy 690TT is highly resistant to stress corrosion cracking, it is more susceptible to wear, which may be caused by the tube support assemblies. Tube wear is a performance degradation mechanism which could lead to collapse of the tubing. This phenomenon can be rapid, and the controlling variables are less commonly understood. Additional tube wall thickness margin has been incorporated as a design feature to address this concern.

Existing guidance based on Pressurized Water Reactor operating experience may not be applicable to the NuScale steam generators. As a result, comprehensive pre-service and in-service inspections will be part of an augmented program to monitor any deviation from expected performance. We concur with the staff assessment that the proposed design and steam generator program, based on applicable industry guidelines, will meet applicable requirements and is acceptable.

The passive DHRS performs a similar function to the auxiliary feedwater system in a current PWR. Two-phase-flow natural circulation is established by boiling in the secondary-side of the steam generators and condensing in the DHRS heat exchangers, which are located outside containment and passively transfer the heat to the reactor pool. This cooling is accomplished by natural circulation primary system flow through the steam generators. As it cools, the primary-coolant water density increases and the level decreases, eventually uncovering the top of the riser, which stops the single phase natural circulation flow. This significantly reduces the efficiency of the steam generators, and the system reaches a self-regulated equilibrium condition based on two-phase-flow natural circulation where steam is condensed on the uncovered portion of the steam generator tubes.

Chapter 15 accident analysis issues are being reviewed to demonstrate acceptability of the DHRS design in satisfying GDC 34 and 35. Verification testing of the system performance has been proposed by the applicant to demonstrate predicted capability for the first-plant to go into operation.

During our Chapter 5 discussions, we learned that unique features of the NuScale design have led to the selection of a ‘radar-based’ technology for measuring water level within the reactor and containment vessels. Water level data provided by these sensors will be used by various safety systems. It is important that qualification of this sensor technology for nuclear applications consider all anticipated environmental conditions (radiation levels, humidity, temperatures, pressures) for their anticipated lifetime and appropriate in-situ calibration methods.

Summary

There are a number of Chapter 4 and Chapter 5 open items related to Chapter 15 accident analysis issues that must be reviewed and resolved in order to demonstrate acceptability of the NuScale reactor design in satisfying GDC 27, GDC 34 and GDC 35. We have not identified any additional major issues at this time for Chapter 4 and Chapter 5.

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 4, 'Reactor'," March 21, 2019 (ML19078A169).

2. NuScale Power, Design Certification Application, Chapter 4, "Reactor," Revision 2, October 30, 2018 (ML18310A325*)*.

3. Advisory Committee on Reactor Safeguards, "Safety Evaluation of the NuScale Power, LLC Topical Report TR-0616-48793, Revision 0, "Nuclear Analysis Codes and Methods Qualification" and Safety Evaluation of the NuScale Power, LLC Topical Report TR-0116-21012, Revision 1, "NuScale Power Critical Heat Flux Correlations," June 15, 2018 (ML18166A303).

4. Advisory Committee on Reactor Safeguards, "Safety Evaluation of the NuScale Power, LLC Topical Report TR-0915-17564-P, Revision 1, "Subchannel Analysis Methodology," September 26, 2018 (ML18270A383).

5. U. S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application - Safety Evaluation With Open Items for Chapter 5, 'Reactor Coolant System and Connecting Systems'," March 21, 2019 (ML19078A060).

6. NuScale Power, Design Certification Application, Chapter 5, "Reactor Coolant Systems and Connecting Systems," Revision 2, October 30, 2018 (ML19078A094).

7. NuScale Power, Technical Report TR-1015-18177, "Pressure and Temperature Limits Methodology," Revision 2, October 25, 2018 (ML18298A304).

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