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

March 26, 2020

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

SUBJECT: SAFETY EVALUATION REPORT OF THE NUSCALE POWER, LLC, TOPICAL REPORT TR-0716-50350, REVISION 1, "ROD EJECTION ACCIDENT METHODOLOGY"

Dear Ms. Doane:

During the 671st meeting of the Advisory Committee on Reactor Safeguards, March 5-6, 2020, we reviewed the NRC staff's safety evaluation (SE) report of NuScale Power, LLC (NuScale), topical report TR-0716-50350, Revision 1, "Rod Ejection Accident Methodology." Our NuScale Subcommittee also reviewed these matters on February 19-20, 2020. During these meetings, we had the benefit of discussions with the staff and representatives of NuScale. We also had the benefit of the referenced documents.

CONCLUSION AND RECOMMENDATION

1. Topical report TR-0716-50350 provides an acceptable methodology for analyses of rod ejection accidents, subject to the limitation of its application to the NuScale reactor design.

2. The staff's SE report should be issued.

BACKGROUND

The NuScale rod ejection accident methodology topical report provides a means to demonstrate compliance with the requirements of Title 10 of the *Code of Federal Regulations,* Part 50 (10 CFR Part 50), Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 28, "Reactivity Limits," which addresses postulated reactivity accidents, including control rod ejection. This postulated accident assumes a sudden ejection of a control rod assembly (CRA) from the core of a critical reactor. The reactor is analyzed at hot zero power and at a range of power levels at any time during an operating cycle.

Ejection of an inserted CRA at hot zero power or low power conditions leads to a rapid increase in power, on the order of several times steady state full power, and it is rapidly terminated by the negative Doppler feedback from U-238 in the fuel, resulting in a localized power pulse. In this scenario the principal effect is fuel rod failure caused by pellet-cladding mechanical interaction, expressed by fuel enthalpy rise.

When the CRA ejection occurs at power conditions, the CRAs are mostly withdrawn. Although this produces a smaller power increase, the system response could lead to over-pressurization of the primary coolant pressure boundary, or to departure from nucleate boiling conditions causing fuel rod failures. The NuScale methodology sets the maximum CRA worth used in the analysis to the power dependent insertion limits, which define allowed control rod patterns and are enforced administratively.

The NuScale rod ejection accident methodology builds on previously submitted topical reports for: nuclear analysis, critical heat flux, and subchannel analysis methods (the VIPRE-01 code); and loss-of-coolant-accident (LOCA) and non-LOCA methodologies (the NRELAP5 code). SIMULATE-3K, a transient 3-D, space-time kinetics code, is used to evaluate the skewed increase in power and localized response of impacted fuel assemblies as the result of a rod ejection accident.

SIMULATE-3K has been benchmarked against the SPERT-III experimental results by the code developer, Studsvik Scandpower. The SPERT-III facility was designed to test and measure reactor kinetic behavior, including fast reactivity transients. The SPERT-III core resembled a small pressurized water reactor similar in physical size to the NuScale reactor design. NuScale performed additional SPERT-III comparisons. The benchmarking also included a reactivity insertion computational benchmark developed by the Nuclear Energy Agency Committee on Reactor Physics. The results of these benchmarks showed excellent agreement for key reactivity and power-related parameters. In their rod ejection accident analyses, NuScale applies uncertainties for the key neutronic parameters of delayed neutron fraction, ejected rod worth, and Doppler and moderator temperature coefficients.

An adiabatic fuel heatup calculation, using total energy input from the SIMULATE-3K transient, is performed to determine if the fuel enthalpy increase and fuel temperature limits (i.e., no melting) are within acceptance criteria. NRELAP5 calculations use output from these 3-D kinetics calculations to confirm that primary coolant pressure boundary limits are not exceeded. The NRELAP5 results are also input to VIPRE-01 to analyze whether critical heat flux limits are exceeded.

DISCUSSION

NuScale has adopted design-specific pellet-cladding mechanical interaction criteria that are within the limits specified in NUREG-0800 Standard Review Plan (SRP), Section 4.2, Appendix B. Specifically, NuScale uses an oxidation criterion of less than 6% of the cladding wall thickness and a fuel enthalpy rise of less than 75 calories/gram. These values are well below cladding failure limits specified in the SRP, as well as those expected to be imposed by DG-1327, “Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents.” This draft guide uses cladding hydrogen content as the metric, whereas the SRP uses the oxide/wall thickness ratio. Application of this methodology shows that the calculated response of the NuScale design to the rod ejection accident design basis scenarios is well within criteria in current regulations and proposed guidance.

The staff has completed a thorough review of the NuScale rod ejection accident methodology. Their evaluation included confirmatory code analyses and audits of code development and application calculations. The staff finds that the NuScale acceptance criteria, the conservatisms used in the approach, and the treatment of input and code parameter assumptions and uncertainties are satisfactory.

In pressurized water reactors, the postulated rod ejection accident is a stylized scenario predicated on rupture of a control rod drive housing and sudden ejection of a control rod assembly from the core of a critical reactor. NuScale did not evaluate the effects of a missile produced by such a rupture on the containment vessel, even though it is only a few feet away. NuScale did not consider this rupture as a potential missile generating source based on the SRP acceptance criteria. NuScale states that they consider potential missiles from piping and valves designed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, and maintained in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, as noncredible. The staff found this approach to be acceptable. While no explicit calculations were performed, the combined probability of a control rod drive mechanism rupture generating a missile and penetrating the containment is likely very low. Therefore, no additional effort is warranted on this topic.

SUMMARY

The Rod Ejection Accident topical report provides an acceptable methodology for analyses of rod ejection accidents, subject to the limitation of its application to the NuScale reactor design. The staff's SE report should be issued.

We are not requesting a formal response from the staff to this letter report.

Member Rempe did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

Matthew W. Sunseri
Chairman

REFERENCES

1. U. S. Nuclear Regulatory Commission, "Safety Evaluation of NuScale Power, LLC Topical Report TR-0716-50350, Revision 1, Rod Ejection Accident Methodology," February 18, 2020 (ML20044E311).

2. NuScale Power, LLC, Topical Report TR-0716-50350, Revision 1, "Rod Ejection Accident Methodology," November 15, 2019 (ML19319C684).

3. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents," November 30, 2016 (ML16124A200).

4. ACRS Letter Report, "Safety Evaluation of the NuScale Power, LLC Topical Report TR-0616-48793, Revision 0, 'Nuclear Analysis Codes and Methods Qualification,' and Safety Evaluation of NuScale Power, LLC Topical Report TR-0116-21012, Revision 1, 'NuScale Power Critical Heat Flux Correlations,'" June 15, 2018 (ML18166A303).

5. ACRS Letter Report, “Safety Evaluation of the NuScale Power, LLC Topical Report TR-0915-17564, Revision 1, ‘Subchannel Analysis Methodology,’” September 26, 2018 (ML18270A383).

6. NuScale Power, LLC, Topical Report TR-0516-49422-P, Revision 1, “Loss-of-Coolant Accident Evaluation Model,” November 30, 2019 (ML19333B884).

7. NuScale Power, LLC, Topical Report TR-0516-49416, Revision 2, “Non-Loss-of-Coolant-Accident Analysis Methodology,” November 30, 2019 (ML19331A895).

8. U.S. Nuclear Regulatory Commission, NUREG-0800, “NUREG-0800 - Chapter 15, Section 15.4.8, Revision 3, Spectrum of Rod Ejection Accidents (PWR),” March 2007 (ML070550014).

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