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

April 21, 2020

The Honorable Kristine L. Svinicki
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SUMMARY REPORT – 671st MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, MARCH 5-6, 2020

Dear Chairman Svinicki:

During its 671st meeting, March 5-6, 2020, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following correspondence:

LETTER REPORT

Letter Report to Kristine L. Svinicki, Chairman, U.S. NRC, from Matthew W, Sunseri, Chairman, ACRS:

- Biennial Review and Evaluation of NRC Safety Research Program, dated April 13, 2020, ADAMS Accession No. ML20100F066

LETTERS

Letters to Margaret M. Doane, Executive Director for Operations (EDO), NRC, from Matthew W. Sunseri, Chairman, ACRS:

- Safety Evaluation Report of the NuScale Power, LLC, Topical Report TR-0716-50350, Revision 1, "Rod Ejection Accident Methodology," dated March 26, 2020, ADAMS Accession No. ML20086Q959

- NuScale Power, LLC, Design Certification Application - Safety Evaluation for Topical Report, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Revision 1, dated March 25, 2020, ADAMS Accession No. ML20085K327

- Safety Evaluation Report for Topical Report TR-0516-49416, Revision 2, "Non-Loss-of-Coolant Accident Analysis Methodology," dated March 25, 2019, ADAMS Accession No. ML20085K048

- NuScale Area of Focus - Helical Tube Steam Generator Design, dated March 24, 2020, ADAMS Accession No. ML20091G387

MEMORANDA

Memoranda to Margaret M. Doane, EDO, NRC, from Scott W. Moore, Executive Director, ACRS:

- "Documentation of Receipt of Applicable Official NRC Notices to the Advisory Committee on Reactor Safeguards for March 2020," dated March 23, 2020, ADAMS Accession No. ML20083H833

- "Regulatory Guides," dated March 23, 2020, ADAMS Accession No. ML20083H690

HIGHLIGHTS OF KEY ISSUES

1. Biennial Review and Evaluation of NRC Safety Research Program

The Committee completed its biennial review and evaluation of safety research sponsored by the NRC. The Committee also reviewed this matter during several information meetings with the NRC Office of Nuclear Regulatory Research (RES) staff on April 4, 2019, July 9, 2019, September 4, 2019, and November 6, 2019.

ACRS reviews of NRC research consider 1997 Commission guidance to examine the need, scope, and balance of the safety research program, and how well RES anticipates research needs and positions itself for the changing environment. This letter report, which was developed using the revised approach the Committee adopted in 2018, emphasizes recommendations to enhance the ability of the RES program to meet current and future agency needs, the prioritization and identification of new research projects, and long-term planning.

Committee Action

The Committee issued a letter report on April 13, 2020, with the following conclusions and recommendations:

a) The current user need process has allowed RES to satisfactorily meet agency near-term needs for regulatory decisions. Efforts to initiate "future-focused" research projects with longer-term horizons will help prepare the agency for upcoming challenges, including the regulation of advanced technologies and the transformation of the agency into a modern, risk-informed regulator.

b) The Committee supports the systematic approach implemented by RES to prioritize research emphasizing "enterprise risk" in project selection, evaluation, and termination. On-going RES efforts to engage other offices are critical for this approach to be successful.

c) As RES continues to leverage resources using international and inter-agency collaborations, it is important that agency priorities be clearly defined and addressed.

d) RES contributions are preparing the agency for anticipated non-light water reactor (LWR) submittals. Reference plant evaluations should provide confidence about the adequacy of selected computational tools and identify any remaining data gaps.

e) The Committee plans to have additional briefings on several RES activities, such as efforts to address the gap created by the unexpected loss of the Halden test reactor, and the selection and progress of future-focused research projects.

2. Safety Evaluation Report of the NuScale Power, LLC, Topical Report TR-0716-50350, Revision 1, "Rod Ejection Accident Methodology"

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 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.

The topical report provides an acceptable methodology for analyses of rod ejection accidents, subject to the limitation of its application to the NuScale reactor design.

Committee Action

The Committee issued a report to the EDO on this topic via letter dated March 26, 2020, with the following conclusion and recommendation:

a) 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.

b) The staff's SE report should be issued.

3. NuScale Power, LLC, Design Certification Application - Safety Evaluation for Topical Report, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Revision 1

The loss-of-coolant accident (LOCA) evaluation model provides a methodology to analyze: the early stages of NuScale LOCA scenarios; the performance of the emergency core cooling system (ECCS); the inadvertent opening of ECCS valves; and peak containment pressure and temperature analysis. The regulatory basis for these analyses is 10 CFR 50.46, and NuScale applies the conservative Title 10 of the *Code of Federal Regulations*, Part 50, Appendix K, "ECCS Evaluation Models," methodology with more restrictive acceptance criteria. As opposed to the conventional LOCA fuel design criteria (e.g., peak cladding temperature limits), NuScale requires that the collapsed liquid level be maintained above the core, which ensures that the fuel is properly cooled during these transients.

The staff has imposed nine limitations and conditions on the use of the topical report for NuScale Power Module (NPM) LOCA evaluations. The NuScale LOCA evaluation model is limited to the evaluation of LOCAs where: the critical heat flux is not exceeded; the collapsed liquid level remains above the top elevation of the core active fuel region for the full spectrum of

break sizes and locations; and the containment peak temperature and pressure remain below the design limits. Application is limited to NRELAP5 version 1.4 with NRELAP5 NPM Model Revision 2 and cannot take credit for cooling by the decay heat removal system.

The LOCA evaluation model topical report, with the limitations and conditions imposed by the staff SE report, provides an acceptable methodology to analyze the early stages of LOCAs prior to long-term cooling in the NPM.

Committee Action

The Committee issued a report to the EDO on this topic via letter dated March 25, 2020, with the following conclusion and recommendation:

a) The LOCA evaluation model topical report, with the limitations and conditions imposed by the staff SE report, provides an acceptable methodology to analyze the early stages of LOCAs prior to long-term cooling in the NPM.

b) The staff's SE report should be issued.

4. Safety Evaluation Report for Topical Report TR-0516-49416, Revision 2, "Non-Loss-of-Coolant Accident Analysis Methodology"

The Non-LOCA Analysis Methodology topical report documents an evaluation model for the analysis of system transient response to non-LOCA initiating events for the NPM. The evaluation model uses a modified version of the RELAP5 computer code, referred to as NRELAP5. It is intended for analysis of anticipated occurrences, infrequent events, and postulated accidents in the NPM. The evaluation model is limited to a short time frame following a design-basis non-LOCA event in which the primary coolant mixture level remains above the top of the core riser, so that primary-side natural circulation is maintained.

The staff concludes that the NRELAP5 validation basis and NPM sensitivity calculation results support the overall conclusion that the NRELAP5 code and the NPM system model are applicable for calculation of non-LOCA system response. The Committee concurs with the staff assessment but is concerned about the screening critical heat flux (CHF) correlation proposed for this evaluation model. While the staff approach is acceptable from a regulatory point of view, the Committee is concerned that it forces the use of a CHF correlation that may be less accurate than other correlations now available. Therefore, adding a condition to the SE report would allow the use of any CHF correlation approved for NPM, as long as its range of applicability covers the transient being analyzed.

Committee Action

The Committee issued a report to the EDO on this topic via letter dated March 25, 2020, with the following conclusions and recommendation:

a) The Non-LOCA analysis methodology topical report, with the limitations and conditions imposed by the staff SE report, provides an acceptable methodology to analyze anticipated occurrences, infrequent events, and postulated accidents for the NPM.

b) The staff should include an additional condition that allows application of this topical report with any critical heat flux correlation approved for use in NPM applications.

c) The staff's SE report should be issued with this additional condition.

5. NuScale Area of Focus - Helical Tube Steam Generator Design

The NuScale NPM steam generator is integral to the upper reactor vessel structure. The helical coil steam generator design is unique, with steam generation inside the tubes and primary system pressure external to the tubes, which introduces different failure modes. Traditional burst analysis that applies to recirculating or once-through steam generators with the primary coolant inside the tubes does not apply. For the NuScale steam generator, a new failure mode would be tube collapse, limiting potential subsequent primary-to-secondary leakage rates compared to a double-ended break. A single steam generator tube rupture has been evaluated in Chapter 15 of the Design Certification Application with acceptable dose consequences. However, if steam generator integrity is not accurately characterized this may not be the limiting event. This also suggests that the estimate of containment bypass under such conditions may be underestimated in the probabilistic risk assessment.

The design and performance of the steam generators have not yet been sufficiently validated because of uncertainties associated with unstable density wave oscillations (DWO) on the steam generator secondary side. Accelerated wear of the alloy 690TT steam generator tubing material is also a potential concern.

Having determined that steam generator integrity is not resolved, NuScale and the staff have proposed the following solutions. The staff has proposed that the steam generator design not receive finality in the NuScale design certification. NuScale has proposed a combined license applicant (COL) item and inspection, test, analysis and acceptance criteria (ITAAC) to address steam generator DWO.

Successful completion of these activities will address our concerns on steam generator performance at the design stage. Some uncertainty will remain until an NPM is built and operated. We look forward to interacting with the staff on the resolution of these items.

Committee Action

The Committee issued a report to the EDO on this topic via letter dated March 24, 2020, with the following conclusions and recommendations:

a) The design and performance of the steam generators have not yet been sufficiently validated because of uncertainties associated with unstable DWO on the steam generator secondary side.

b) Accelerated wear of the alloy 690TT steam generator tubing material is a potential concern.

c) Having determined that steam generator integrity is not resolved, NuScale and the staff have proposed the following solutions:

   - The staff has proposed that the steam generator design not receive finality in the NuScale design certification.
   - NuScale has proposed a combined license (COL) item and Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) to address steam generator DWO.

d) Successful completion of these activities will address our concerns on steam generator performance at the design stage. Some uncertainty will remain until a NPM is built and operated.

AMENDMENT TO BYLAWS TO CLARIFY USE OF VIRTUAL MEETINGS

The Committee discussed the potential need to clarify the ACRS bylaws to specifically allow the Full Committee and Subcommittee meetings to be conducted virtually. This was prompted by the potential for use of such meetings due to the COVID-19 pandemic. Currently the bylaws do not mention use of virtual meetings. The Committee voted to authorize the ACRS Committee leadership to work on a proposed amendment to the bylaws, if needed, in accordance with Section 14, “Amendment,” to clarify the potential for using virtual meetings to carry out the Committee’s duties. The Committee plans to discuss this bylaw issue at the April meeting.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

The Committee considered the correspondence from the Director of the Office of Nuclear Reactor Regulation (NRR), dated January 27, 2020, ADAMS Accession No. ML20006D944, in response to the Committee’s letter dated December 20, 2019, ADAMS Accession No. ML19354A031. The topic was the 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. The Committee accepted the staff’s response.

SCHEDULED TOPICS FOR THE 672nd ACRS MEETING

The following topics were placed on the agenda for the 672nd ACRS meeting which is scheduled for April 8-11, 2020:

a) Surry Power Station Subsequent License Renewal Application Review

b) Various NuScale Design Certification Application Review Topics

c) Kairos Advanced Reactor Design Topical Report Subjects

Sincerely,

/RA/

Matthew W. Sunseri,
Chairman

April 21, 2020

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