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

July 12, 2019

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

SUBJECT: SUMMARY REPORT – 664th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JUNE 5 - 7, 2019

Dear Chairman:

During its 664th meeting, June 5 - 7, 2019, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following correspondence:

REPORT

Letter Report to Kristine L. Svinicki, Chairman, NRC, from Peter C. Riccardella, Chairman, ACRS:

- "Review of Nuclear Energy Institute (NEI) 96-07, Appendix D, 'Supplemental Guidance for Application of 10 CFR 50.59 to Digital Modifications,' dated November 2019, and the NRC's Associated Draft Revision 2 to Regulatory Guide 1.187, 'Guidance for Implementation of 10 CFR 50.59 Changes, Tests, and Experiments'," dated June 20, 2019, Agencywide Documents Access and Management System (ADAMS) Accession No. ML19171A323.

LETTERS

Letters to Margaret M. Doane, Executive Director for Operations (EDO), NRC, from Peter C. Riccardella, Chairman, ACRS:

- "Interim Letter: NuScale Design Certification Application Chapters 3.9.2, 14, 19, and 21," dated June 19, 2019, ADAMS Accession No. ML19170A381.

- "Interim Letter: NuScale Design Certification Application Chapters 2 and 17," dated June 27, 2019, ADAMS Accession No. ML19171A350.

MEMORANDA

Memoranda to Margaret M. Doane, Executive Director for Operations, NRC, from Andrea D. Veil, Executive Director, ACRS:

- “Documentation of Receipt of Applicable Official NRC Notices to the Advisory Committee on Reactor Safeguards for June 2019,” dated June 27, 2019, ADAMS Accession No. ML19178A295.

- “Regulatory Guide,” dated June 27, 2019, ADAMS Accession No. ML19178A315, regarding no review of RG 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” Revision 4

HIGHLIGHTS OF KEY ISSUES

1. Letter Report regarding review of Nuclear Energy Institute (NEI) 96-07, Appendix D, “Supplemental Guidance for Application of 10 CFR 50.59 to Digital Modifications,” dated November 2019, and the NRC’s Associated Draft Revision 2 to Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59 Changes, Tests, and Experiments”

Nuclear Energy Institute 96-07, Appendix D, is intended to assist licensees in the performance of 10 CFR 50.59 reviews of activities involving digital modifications. Draft Revision 2 to Regulatory Guide 1.187, endorses the current version of NEI 96-07, Appendix D, with exceptions and clarifications as providing an acceptable approach for the application of 10 CFR 50.59 guidance when conducting DI&C modifications.

There remains one area of disagreement between the NRC staff and NEI related to the interpretation of 10 CFR 50.59, Section (c)(2)(vi), “Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.” The phrase ‘different result’ is interpreted differently by NRC staff and NEI. The staff lays out its objections to the NEI 96-07, Appendix D, interpretation, in Section C.2 of the Regulatory Guide, which should be revised for clarity. The essence of the staff position, based on rule language, is that ‘a different result’ means that the evaluation should determine the impact of the SSC malfunction anywhere in the UFSAR. NEI’s position, based on language in the original Statement of Considerations to 10 CFR 50.59 and the definitions in NEI 96-07, is that ‘a different result’ means that the replacement SSC changes the results of a safety analysis in the UFSAR. To treat the 10 CFR 50.59 requirements fairly across plants with detailed UFSARs and those with less extensive UFSARs, NEI stated that a ‘malfunction’ is defined as failure to perform a design function; that although specific SSCs and their malfunctions may differ across individual plants, design functions do not; and, finally, that the replacement SSC should be evaluated based on the design functions it could affect, regardless of whether the SSC is specifically discussed in the safety analyses. This is an important concept that should be considered in the staff’s approach as well. Draft Regulatory Guide 1.187, Revision 2, endorsing NEI 96-07, Appendix D, was issued for public comment on May 30, 2019, with the above issue unresolved.

Committee Action

The Committee issued a report to the NRC Chairman with the following conclusions and recommendations.

- o Guidance for applying 10 CFR 50.59 to DI&C systems has been needed. This stems from the inherently different failure characteristics of systems that include DI&C equipment and from the unique and far-reaching potential impacts of DI&C system common-cause events.

- o Draft Revision 2 to Regulatory Guide 1.187, that endorses NEI 96-07, Appendix D, with exceptions and clarifications, provides an acceptable and timely approach for applying 10 CFR 50.59 guidance when conducting DI&C modifications.

- o A staff exception to NEI 96-07, Appendix D, requires consideration of more than the safety analysis within the updated final safety analysis report (UFSAR) and, thereby constrains its application. There is an opportunity for expanding the use of 10 CFR 50.59 for DI&C modifications by more clearly identifying the significance of different results caused by a malfunction of a Structure, System, and Component (SSC) important to safety as specified in Criterion 6. The use of risk-informed or other methods should be considered. This is a longer-term issue and may require a rule change.

- o The staff should provide final Revision 2 to Regulatory Guide 1.187 for our review following resolution of public comments.

2. <u>Interim Letter: NuScale Design Certification Application Chapters 3.9.2, 14, 19, and 21</u>

The Committee met with representatives of the NRC staff and NuScale to review the NuScale Small Modular Reactor Design Certification Application (DCA) 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," and associated staff safety evaluation reports with open items.

Chapter 3, Section 3.9.2 of the DCA describes 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.

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 Flow-induced vibration (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.

Chapter 14 describes the initial test program that consists of a series of preoperational and startup tests conducted by the startup organization. Preoperational testing is conducted for each NuScale Power Module (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. Chapter

14 also describes the inspections, tests, analyses and acceptance criteria that will be used to verify that the licensed plant is built in accordance with the design and license requirements.

Chapter 19 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. 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.

Chapter 19 also describes the analysis of the prevention and the mitigation of severe accidents. This chapter 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.

Chapter 21 contains a description 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.

Committee Action

The Committee issued a report to the EDO on these Chapters and associated safety evaluation with open items, dated June 19, 2019, with the following conclusions and recommendations:

- 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),
- The ACRS has not identified any major issues at this time for Chapter 3, Section 3.9.2, and Chapters 14, 19 and 21, and
- To help identify risk insights in this unique design, there are technical issues in the probabilistic risk assessment (PRA) that merit further consideration.

3. Interim Letter: NuScale Design Certification Application Chapters 2 and 17

The Committee met with representatives of the NRC staff and NuScale to review the NuScale Small Modular Reactor Design Certification Application (DCA) Chapter 2, "Site Characteristics and Site Parameters," and Chapter 17, "Quality Assurance and Reliability Insurance," and associated staff safety evaluation reports with open items.

Chapter 2 describes generic site parameters and characteristics for which the design is approved and certified.

Chapter 17 describes the quality assurance and reliability insurance measures to be undertaken to ensure the plant meets all specified requirements.

The Committee recommended, in its letter dated June 27, 2019, that the staff's review process prior to design certification should identify those unverified design assumptions that are important to the reasonable assurance finding. This will help ensure that required closure prior to operability of the affected structure, system or component is identified and included as part of the design certification licensing basis. It is important that this process is not limited to items covered by the quality assurance program.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee considered the letter from the Executive Director for Operations dated December 15, 2016, ADAMS Accession No. ML16343A131, as supplemented by letter from the Director of the Office of Nuclear Reactor Regulation dated April 9, 2019, ADAMS Accession No. ML19053A532, regarding the draft final rule on mitigation of beyond-design-basis events and associated regulatory guidance. The Committee understands the staff response and looks forward to interaction with the staff, as warranted, on such matters.

SCHEDULED TOPICS FOR THE 665th ACRS MEETING

The following topics are on the agenda for the 665th ACRS meeting scheduled for July 10 - 12, 2019:

- NuScale Design Certification Application Chapters 3, 6, 15, and 20 and the Stability Topical Report

Sincerely,

/RA/

Peter C. Riccardella,
Chairman

July 12, 2019

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