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

December 16, 2019

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

SUBJECT: SUMMARY REPORT – 668th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, NOVEMBER 6-8, 2019

Dear Chairman Svinicki:

During its 668th meeting, November 6-8, 2019, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following correspondence:

LETTERS

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

- Interim Letter: The NRC Staff's Safety Evaluation Report with No Open Items for Chapters 8 and 18 and the Advanced Accumulator Topical Report Related to the Certification of the US-APWR Design, dated November 26, 2019, ADAMS Accession No. ML19331A036

- NUREG/KM-0013, "Credibility Assessment Framework for Critical Boiling Transition Models – A Generic Safety Case to Determine the Credibility of Critical Heat Flux and Critical Power Models, Draft Report for Comment," dated November 26, 2019, ADAMS Accession No. ML19331A067

- Safety Evaluation for Brunswick Steam Electric Plant Units 1 and 2 to Support Review of the License Amendment Request Regarding Application of FRAMATOME Methodologies for Transition to ATRIUM-11 Fuel, dated November 26, 2019, ADAMS Accession No. ML19331A112

- Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99, dated November 27, 2019, ADAMS Accession No. ML19331A231

MEMORANDA

Memorandum to Margaret M. Doane, Executive Director for Operations, NRC, from Scott W. Moore, Executive Director, ACRS:

- "Documentation of Receipt of Applicable Official NRC Notices to the Advisory Committee on Reactor Safeguards for October 2019," dated December 10, 2019, ADAMS Accession No. ML19346F107.

Memorandum to Ho K. Nieh, Director, Office of Nuclear Reactor Regulation (NRR), NRC, from Scott W. Moore, Executive Director, ACRS:

- ACRS Review of NuScale Power, LLC, Design Certification Application – Safety Evaluation with No Open Items for Chapters 5 and 21, dated December 10, 2019, ADAMS Accession No. ML19337B671.

HIGHLIGHTS OF KEY ISSUES

1. Interim Letter: The NRC Staff's Safety Evaluation Report with No Open Items for Chapters 8 and 18 and the Advanced Accumulator Topical Report Related to the Certification of the US-APWR Design

The Committee met with representatives of the NRC staff and Mitsubishi Heavy Industries, Ltd (MHI), to discuss the design certification application and associated NRC safety evaluation reports. The specific topics discussed included the Design Control Document (DCD), Chapter 8, "Electric Power," DCD, Chapter 18, "Human Factors Engineering," Topical Report MUAP-07001, "The Advanced Accumulator," and associated staff safety evaluation reports (SERs).

Regarding Chapter 8 issues, safety-related emergency alternating current (AC) power for the US-APWR is supplied by four gas turbine generators (GTGs). The use of GTGs for safety-related AC power is a departure from the historical use of diesel generators. An issue raised during our earlier subcommittee meetings was the need to confirm MHI's reliability estimates for the GTG emergency power systems, accounting for support equipment. MHI presented the results of their qualification testing program. This program, among other attributes, performed 150 start tests without failure to demonstrate a reliability criterion of 0.975 with 95 percent confidence. Successful completion of the testing program presented by MHI resolves our GTG reliability concern.

Regarding Chapter 18 issues, MHI has modified a predecessor Japanese control room design and associated procedures to be consistent with U.S. operational practice. The approach is thorough and makes extensive use of experienced U.S. crews operating a full-scale plant simulator. For the remainder of the HFE tasks, implementation plans and Inspections, Tests, Analyses, and Acceptance Criteria have been developed.

Regarding the advanced accumulator (ACC), MHI initially qualified the ACC using a combination of testing and computational fluid dynamics (CFD) analysis. The Committee reviewed the previous qualification analysis that used half-scale testing with extrapolation to full scale using CFD. Although the methodology was acceptable, the Committee concurred with the staff's recommendations to increase the uncertainties that are used in loss-of-coolant accident (LOCA) analyses for the high-flow and low-flow injection regimes. The increased uncertainties account for the use of CFD analysis models to extend the half-scale test results to predict full-scale accumulator performance. MHI has subsequently performed full-scale testing of the ACC to justify removing scaling uncertainties. The characteristic equations developed from the full-scale test facility are applicable to the full-scale accumulator with the remaining uncertainties and bias described in their updated report. MHI confirmed that the Chapter 15 LOCA analysis will be rerun with the new correlation equations.

Committee Action

The Committee issued a report to the Executive Director for Operations (EDO) on November 26, 2019, with the following conclusions and recommendation:

a) The Committee's review of the SERs for Chapters 8 and 18 did not identify any safety issues that would preclude issuance of a design certification at this stage of our review. The Committee will continue to consider integral effects of system interactions as we complete our final review.

b) The Committee's review of the SER for the topical report on the advanced accumulator did not identify any safety issues.

c) The SERs should be issued.

2. NUREG/KM-0013, "Credibility Assessment Framework for Critical Boiling Transition Models – A Generic Safety Case to Determine the Credibility of Critical Heat Flux and Critical Power Models, Draft Report for Comment"

NUREG/KM-0013 offers a systematic and auditable approach to the review of data-driven models, also referred to as correlations. Such models are used across the industry to apply results from separate effects experiments to licensing calculations.

The credibility assessment framework considers three high-level decisions that assess: the goodness of the experimental data, the adequacy of the model generation, and the validation of the final correlation and its uncertainties. The framework allows the staff to systematically break down their review into small logical segments that allow efficient evaluation of these high-level decisions. One worthy example is how the methodology allows the objective definition of subjective terms such as "safe." In the process, the evaluation documents the review in an auditable way that can be easily traced and updated for future applications.

Committee Action

The Committee issued a report to the EDO on this topic via letter dated November 26, 2019, with the following conclusion and recommendation:

a) The credibility assessment framework documented in NUREG/KM-0013 is an innovative technical approach to review the adequacy of data-driven models.

b) Extending this framework to other data-driven model applications should be explored.

3. Safety Evaluation for Brunswick Steam Electric Plant Units 1 and 2 to Support Review of the License Amendment Request Regarding Application of FRAMATOME Methodologies for Transition to ATRIUM-11 Fuel

The licensee for Brunswick Steam Electric Plant Units 1 and 2 (BSEP) submitted a license amendment request (LAR) to operate with ATRIUM™ 11 fuel and adopt Framatome methods for core reload evaluations. This LAR is the first request to apply these methodologies. The Committee met with members of the NRC and licensee staff during the meeting.

The BSEP LAR contains several changes to the technical specifications and core operating limits report (COLR), primarily by incorporating references to these new Framatome methods. Therefore, future core reload analyses can properly reference these methods as approved.

The staff review concludes that the LAR provides an acceptable implementation of the previously approved generic analysis methods, including: the fuel assembly design and chromia-doped fuel property correlations; anticipated operational occurrences; anticipated transient without scram overpressure; and control rod drop accidents. In their review, the staff confirmed that limitations and conditions for these methods were addressed appropriately in the LAR application.

Committee Action

The Committee issued a report to the EDO on this topic via letter dated November 26, 2019, with the following conclusion and recommendation:

a) The Framatome core reload analysis methodology is acceptable for use in BSEP Units 1 and 2 licensing applications that incorporate ATRIUM™11 fuel in their currently approved extended power-flow operating domain.

b) The LAR should be approved, and the SE should be issued.

4. Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99

The Committee met with representatives of the NRC staff and the Electric Power Research Institute (EPRI) during this session.

Regulatory Guide 1.99, “Radiation Embrittlement of Reactor Vessel Materials,” Revision 2, describes methods that may be used to predict the effects of radiation embrittlement of reactor pressure vessels (RPVs). Specifically, neutron irradiation of the RPV steel results in material property changes making the steel more brittle and potentially susceptible to rapid failure under high-stress conditions. This effect increases with neutron fluence. The embrittlement of RPV steels can pose a safety challenge that impacts operational pressure-temperature limits. An embrittlement trend correlation is used to calculate the shift in reference nil-ductility temperature (ΔRT_{NDT}) as a function of fast neutron fluence.

The most recent revision of this guide (Revision 2) was published in 1988. At that time, the number of data points available for development of the correlation was 177. It was expected at the time of publication that the regulatory guide would be updated and refined as more material data became available. The current data base now contains approximately 1900 data points.

The staff recently evaluated predictions using RG 1.99, Revision 2 for higher fluences that will be experienced during subsequent license renewal (SLR) periods. Results demonstrate that the correlation in this RG introduces significant errors that are non-conservative at higher fluence. The adoption of new guidance regarding prediction of the effects of embrittlement may have significant impact on all operating Pressurized Water Reactors (PWRs) that previously used RG 1.99, Revision 2 to develop Pressure-Temperature (P-T) curves, Low Temperature Overpressure Protection setpoints, and pressurized thermal shock (PTS) limits.

In response to the increasing number of plants that have applied for SLR, the industry has embarked on an extensive program of data gathering at high fluence. It is expected that actual plant data will be available for verification of embrittlement trends well before the existing PWRs will require it for extended operation.

The Committee discussed several technical aspects of this issues as detailed in the referenced letter. The Committee looks forward to reviewing the updated RG and implementation plan.

Committee Action

The Committee issued a report to the EDO on this issue via letter dated November 27, 2019, with the following conclusion and recommendations:

a) The embrittlement trend correlation (ETC) in RG 1.99, Revision 2 (the RG) has a number of deficiencies, the most significant of which is increasing error beyond a fluence of 6×10^{19} n/cm^2 (E >1 MeV).

b) The American Society for Testing and Materials (ASTM) Subcommittee E10.02, Behavior and Use of Nuclear Structural Material, has performed an extensive review of several ETCs. It concluded that the correlation in ASTM E900-15, that is based on a much more extensive database, overcomes the deficiencies in the RG and provides the best fit at higher fluences.

c) A staff working group has been established and has identified a path forward for addressing this issue.

d) A staff oversight group has also been established to guide the implementation of a revision to the RG to correct its deficiencies. This group should consider each plant's situation to eliminate unnecessary burden on plants for which RPV limits are not challenged.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee considered the letter from a cognizant branch chief from the Office of Nuclear Reactor Regulation, dated March 21, 2019, ADAMS Accession No. ML19066A297, in response to the Committee's letter dated February 22, 2019, ADAMS Accession No. ML19057A018. The topic was the safety evaluation for topical report ANP-10332P, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios." The Committee accepted the response which agreed that the SER should be issued.

- The Committee also considered the letter from the Director, Office of New Reactors, dated September 13, 2019, ADAMS Accession No. ML19241A273, in response to the Committee's letter dated August 2, 2019, ADAMS Accession No. ML19204A278. The topic was the ACRS's interim letter on NuScale design certification application and associated staff SERs for Chapters 3, 6, 15 and 20. The Committee understands the staff's response and looks forward to further reviewing important safety-significant issues, especially those associated with Chapter 15 regarding design basis accident analyses, during the Phase 5 review. One specific topic that should be addressed is potential boron dilution and return to criticality. The Committee looks forward to addressing the areas of focus that have been identified by the Committee during the Phase 5 review (see below).

NUSCALE PHASE 5 DISCUSSIONS

As documented in the memorandum from Scott Moore, ACRS Executive Director, to Ho Nieh, Director, NRR, the Committee decided that no further briefing by the staff is needed to support the Committee's Phase 5 review as it pertains to Chapters 5 and 21 of the NuScale design certification application. The Committee decided that no further briefing by the staff is needed to support the Committee's review as it pertains to Chapters 5 and 21 of the NuScale design certification application. The Committee plans to complete its review of the reactor coolant system and its connecting systems as part of its Emergency Core Cooling System (ECCS) and steam generator focus area reviews. In addition, the Committee plans to complete its review of multi-module risk as part of its remaining chapter and focus area reviews. The Committee will inform the Commission of its finding in this matter relative to the requirements of 10 CFR 52.53 in a subsequent letter.

The Committee identified its approach to the Phase 5 review in its letter dated September 25, 2019 (ADAMS Accession No. ML19269B682).

SCHEDULED TOPICS FOR THE 669th ACRS MEETING

The following topics were placed on the agenda for the 669th ACRS meeting which was scheduled for December 4-7, 2019:

- NuScale Source Term Topical Report Methodology and the Source Term Focus Area as part of the Phase 5 review

- Peach Bottom Subsequent License Renewal Application Review

- Susquehanna ATRIUM-11 fuel transition

- General Electric/Global Nuclear Fuels Control Rod Drop Accident Methodology

- Further discussion of various safety evaluation reports related to the NuScale design certification application review

Sincerely,

/RA/

Peter C. Riccardella,
Chairman

December 16, 2019

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Accession No: ML19351F531 **Publicly Available (Y/N):** _Y___ **Sensitive (Y/N):** N
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