SCHEDULING NOTE

Title: **MEETING WITH ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (Public)**

Purpose: Meeting with the NRC's independent Advisory Committee on Reactor Safeguards to provide their views to the Commission on issues recently reviewed by the Committee.

Scheduled: **December 6, 2019**
10:00 a.m.

Duration: Approx. 2 hours

Location: Commissioners' Conference Room, 1st fl OWFN

Participants: **Presentation**

<u>ACRS Members</u> **50 mins.***

Peter Riccardella, Chairman, Advisory Committee on Reactor Safeguards (ACRS)

- Overview
- Transformation

Walter Kirchner, Member, ACRS

- NuScale Design Certification Application Review

Dennis Bley, Member, ACRS

- Advanced Reactor Siting
- Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors
- Advanced Reactor Computer Codes

Joy Rempe, Member, ACRS

- Assessment of the Quality of Selected NRC Research Projects

Commission Q & A **40 mins.**

Discussion – Wrap-up **5 mins.**

*For presentation only and does not include time for Commission Q & A's

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

U.S. NUCLEAR REGULATORY COMMISSION MEETING WITH THE ACRS

December 6, 2019

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FINAL: 10/29/19

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Documents:

Background information due: 11/22/19
Slides due: 11/26/19

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

COMMISSION MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

December 6, 2019

Agenda

- Pete Riccardella, Chairman, ACRS
  - Overview and Transformation
- Walter Kirchner, Member, ACRS
  - NuScale Design Certification Application Review
- Dennis Bley, Member, ACRS
  - Advanced Reactor Siting; Technology-inclusive, Risk-informed, and Performance-based Approach; Advanced Reactor Computer Codes
- Joy Rempe, Member-at-Large, ACRS
  - Assessment of the Quality of Selected NRC Research Projects

Accomplishments

Issued 31 reports since the last meeting with the Commission in December 2018:

- NuScale Design Certification Application (DCA)
  - Safety Evaluation Reports (SERs) with Open Items (8)
  - NuScale Topical Reports (2)
  - Focus Area Review Approach (1)
- ACRS Activities to Support NRC Transformation

Accomplishments (Cont'd)

- Advanced Reactor Topics (3)
- License and Design Certification Renewals (5)
- Vendor Topical Reports (4)
- Other Topics
  - 10 CFR 50.59 for DI&C
  - Expanded Power-to-Flow Domain Application Reviews
  - Non-power, Production, or Utilization Facility (NPUF) Rulemaking
  - TVA Clinch River Early Site Permit
  - Reactor Vessel Embrittlement Technical Letter Report (Re: Regulatory Guide 1.99)
  - Quality Review of Selected RES Projects

ACRS Transformation

Committee engaged in several activities to assess ACRS role in a transformed agency:

- Briefed by senior NRC staff
- Conducted ACRS retreats and discussed at Committee meetings
- Solicited input from the EDO, current and past Commissioners
- Reviewed relevant agency transformation documents

Conclusions and Proposed Actions

- ACRS reviews provide integrating perspective and increase quality and rigor
- Moving forward
  - Prioritize reviews based on risk significance and agency transformation priorities
  - Stay abreast of staff transformation initiatives and continue to contribute
  - Improve operational efficiency
- No need for rule changes to implement these actions

Actions Already Underway

- Established prioritization criteria for Committee review topics
- Developed (with staff) a more effective process for NuScale DCA Phase 5 review
  - Focused on risk-significant, cross-cutting issues instead of another chapter-by-chapter review
- Eliminated reviews of some routine, low priority items
- Implemented process improvements to enhance operational efficiency

Summary

- ACRS performs independent, integrated, multi-discipline reviews
- Prioritization of future reviews will focus on those with the most impact and value to the Commission
- Membership with expertise covering the breadth of risk-significant issues is mission-critical

NuScale Design Certification Application (DCA) Review

Walter Kirchner, Chair, ACRS NuScale Subcommittee

NuScale DCA

- NuScale Power Modules (NPM)
  - Small modular, natural circulation PWR
  - 160 MWt/50 MWe per module
  - Each NPM composed of reactor core, pressurizer, and two helical steam generators integral to a reactor vessel and enclosed in a high-strength steel containment vessel

NuScale DCA (Cont'd)

- Core contains 37 ~half-length 17 x 17 PWR fuel assemblies
- Each NPM has a dedicated, passive emergency core cooling system (ECCS) and decay heat removal system (DHRS), not reliant on electrical power

NuScale DCA (Cont'd)

- Reactor Building
  - NPMs largely immersed in common pool of water
  - Pool serves as passive ultimate heat sink for cooling during design basis events (DBEs) and beyond DBEs (BDBEs)
  - Common pool for refueling and spent fuel storage

NuScale Review Status

- Met Phase 3 milestone of August 27, 2019
- Issued 7 Interim Chapter Letter Reports (for 21 Chapters)
- Issued 8 Topical Letter Reports
- Four Topical Reports remain to be reviewed

Phase 5 Review

- Cross-cutting "Areas of Focus" review proposed for Phase 5 based on lessons learned from past DCA reviews
- Consistent with NRC's strategy for transforming to more risk-informed, performance-based, safety-focused reviews
- In-depth review of matters that are inherently cross-cutting regarding integrated system safety performance

Phase 5 Review (Cont'd)

- ACRS chapter lead will perform detailed chapter review and document for completeness
- Lead for chapter will make recommendation to Full Committee if briefing is needed, or to include items in a focus area review

Phase 5 Review (Cont'd)

- The currently identified focus area reviews include:
  - ECCS and Valve Performance
  - Helical-Tube Steam Generator Design
  - Boron Dilution and Return to Criticality
  - Source Term
  - Probabilistic Risk Assessment

Phase 5 Review (Cont'd)

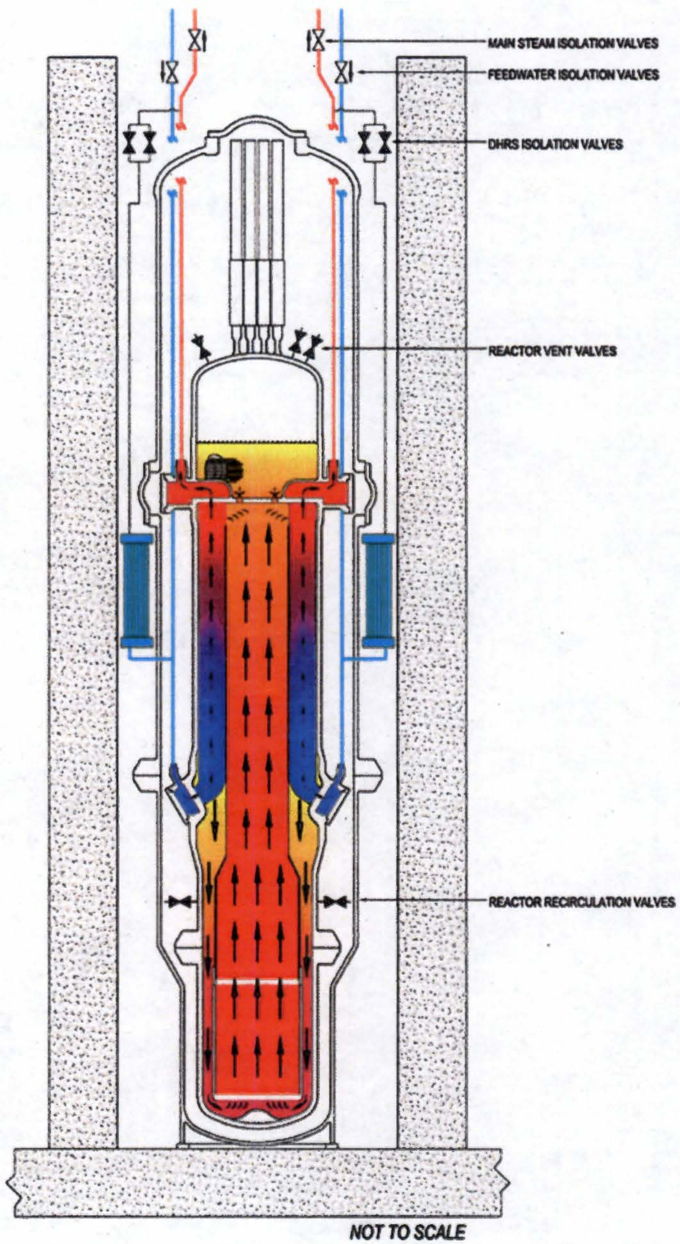
- This is a departure from past reviews of design certification applications (chapter by chapter)
- Less resource intensive for staff and applicant; more effective safety focus
- EDO and staff expressed favorable feedback

Phase 5 Review Status

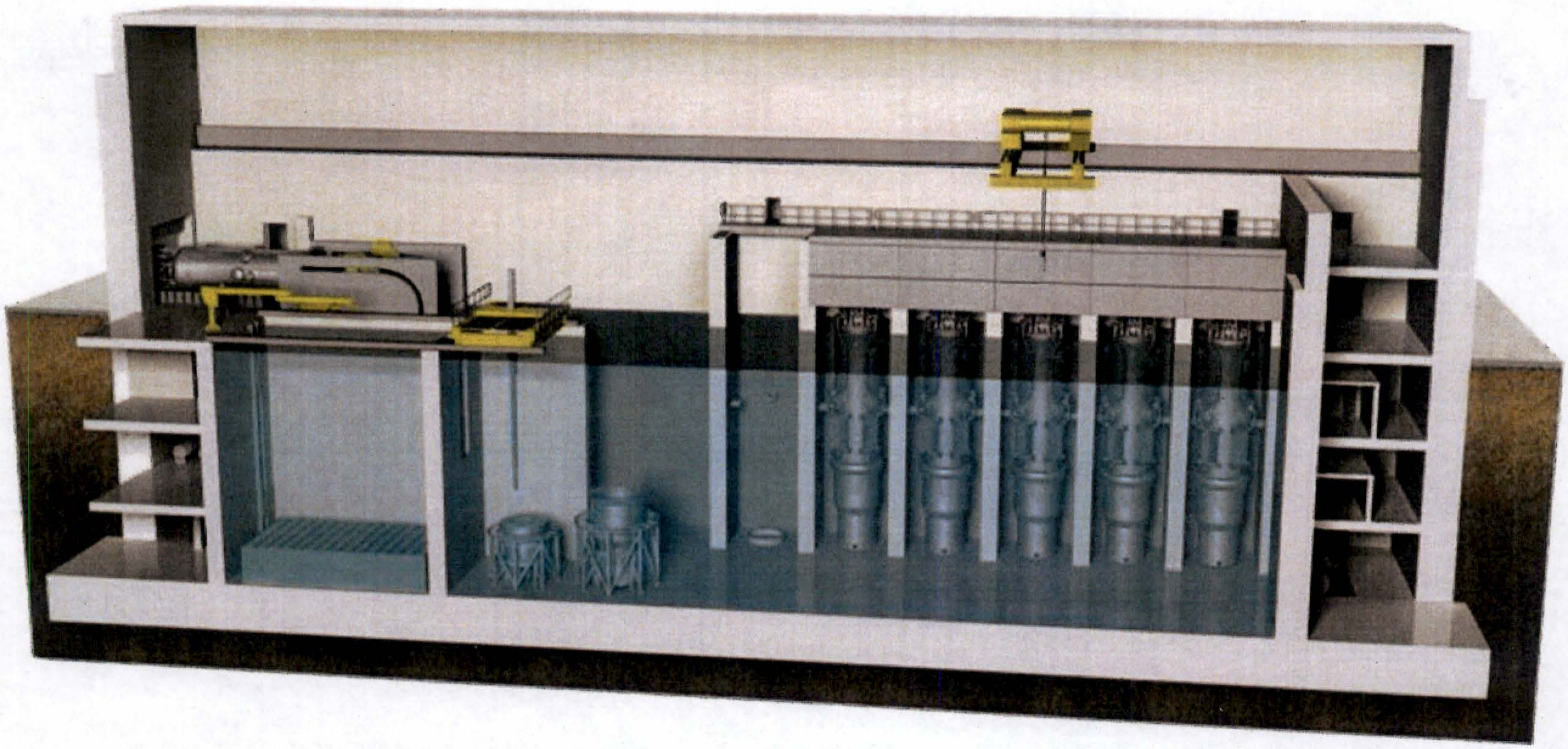
- All SERs with no open items due on December 12, 2019
- Six SER chapters have been reviewed by chapter leads and determined not to require a briefing
- Chapter 15 SER briefing scheduled for February/March 2020
- Focus area briefing schedule being negotiated with staff for early 2020
- Working with staff to meet June 23, 2020 target milestone

NuScale Backup Slides

NuScale Power Module



NuScale Reactor Building



Three Reports on the Staff's Vision and Strategy for Review of Non-LWR Applications

Dennis Bley, Chair
ACRS Future Plant Designs Subcommittee

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Near-Term Implementation Action Plan

Strategy 1 Knowledge, Skills, Capacity	Strategy 2 Computer Codes	Strategy 3 Flexible Review Process	Strategy 4 Industry Codes and Standards	Strategy 5 Technology-Inclusive Policy Issues	Strategy 6 Communication
ORNL Molten Salt Reactor Training	Identification/A ssessment of available tools	Regulatory Roadmap	ASME BPVC Sect. III Div 5	Siting near densely populated Areas	NRC DPE Workshops
Knowledge Management		Prototype Guidance	ANS Standards 20.1, 20.2, 30.2,54.1	Insurance and Liability	Periodic Stakeholder Meetings
Competency Modeling		Non-LWR Design Criteria	Non-LWR PRA Standard	Consequence-Based Security	NRC DOE GAIN MOU
		Licensing Modernization Project • LBE selection • Defense in Depth • PRA Approach • SSC Safety Classification		EP for SMRs and ONTs	International Coordination
				Functional Containment Performance Criteria	

Licensing Modernization Project (LMP)

- LMP: Technology-inclusive, risk-informed, and performance-based approach to inform the content of applications for licenses, certifications, and approvals for non-LWR reactors – gathered in NEI 18-04
- DG-1353 endorses with clarifications, principles and methodology of NEI 18-04
- Proposed approach neither exempts any design from existing regulations nor addresses all regulations applicable to nuclear power plants

Objectives – LBEs, SSCs, DID

- Identify Licensing Basis Events (LBEs)
  - Defined by scenarios developed in the PRA
  - Tested against frequency-consequence goals in NEI 18-04
  - Total integrated risk must meet integrated goals
  - Includes AOOs, DBEs, BDBEs now defined objectively by PRA frequency results

Objectives (Cont'd)

- Classify Structures, Systems, and Components (SSCs)
  - Paper extends and makes operational concepts expressed earlier
  - SSCs selected from important risk contributors in PRA
  - Special treatment assigned based on importance to risk
- Defense in Depth (DID)
  - Operational structure for evaluation of DID
  - Uses techniques to evaluate plant capabilities and programmatic controls
  - No reliance on a single element of design/program

ACRS Findings and Recommendations

1. Next evolution of a licensing approach in development for thirty years
2. Three objectives: select LBEs, classify SSCs, assess adequacy of DID
3. Recommend adoption of approach
4. Guidance in DG-1353 is adequate to support implementation, except source term
5. DG-1353 should be issued for comment

Population-Related Siting Considerations

Existing Regulatory Framework

- Exclusion area (EA), low population zone (LPZ), and population center distance (PCD)
- EA and LPZ boundaries set by dose limits of 25 Rem (2 hours/entire cloud)
- PCD 1.33 times the radius of the LPZ from boundary of any densely populated center >25,000 people

Existing Regulatory Framework

RG 4.17 written for large LWRs:

- A reactor should be located so, at the time of initial plant approval and within about 5 years thereafter, the population density, over any radial distance out to 20 miles does not exceed 500 persons per square mile (ppsm)
- A reactor should not be located at a site where the population density is well in excess of this value

Options Evaluated

- Option 1 – Status quo
- Option 2 – Scaling source term with power
- Option 3 – Dose-based
- Option 4 – Develop societal risk measure

Option 3 Dose-Based

New guidance in RG 4.17 for small modular reactors (SMRs) and microreactors

- Density of 500 ppsm assessed to distance equal to twice the distance at which a hypothetical individual could receive 1 rem over 1 month after hypothetical design accident
- Recommended

ACRS Findings and Recommendations

1. ACRS agrees that Option 3 is reasonable, however paper is short on implementation details
2. These details should be provided in RG 4.17 with illustrative examples

Advanced Computer Code Evaluations

We have reviewed available volumes of the Strategy 2 Report on Codes for:

- DBE Analysis (systems analysis)
- Fuel Performance Analysis
- Severe Accident Progression, Source Term, and Consequence Analysis

ACRS Findings and Recommendations

1. Approach supports readiness of NRC staff to review non-LWR reactor applications and can help staff understand new designs
2. Tools for staff confirmatory analysis should be as independent as practical and validated
3. Staff needs to become familiar with applicant codes to support timely reviews

ACRS Findings and Recommendations (Cont'd)

4. The overview report should be revised to better explain how the approach integrates the evaluations using a coherent strategy
   - Four principles should underlie the strategy: simplicity, completeness, working the problem backwards starting with source term (ST), and scaling down the level of effort as hazard decreases
5. The staff should perform pilot studies using relatively mature designs to illustrate how the analysis should proceed

Assessment of the Quality of Selected NRC Research Projects

Joy Rempe, Chair
ACRS Safety Research Subcommittee

Background

- Throughout its history, an essential ACRS activity is reviewing NRC-sponsored research
- This activity includes reviews of:
  - Research conducted in support of specific regulatory activities
  - Important ongoing agency research
  - NRC safety research program
  - The quality of specific research projects

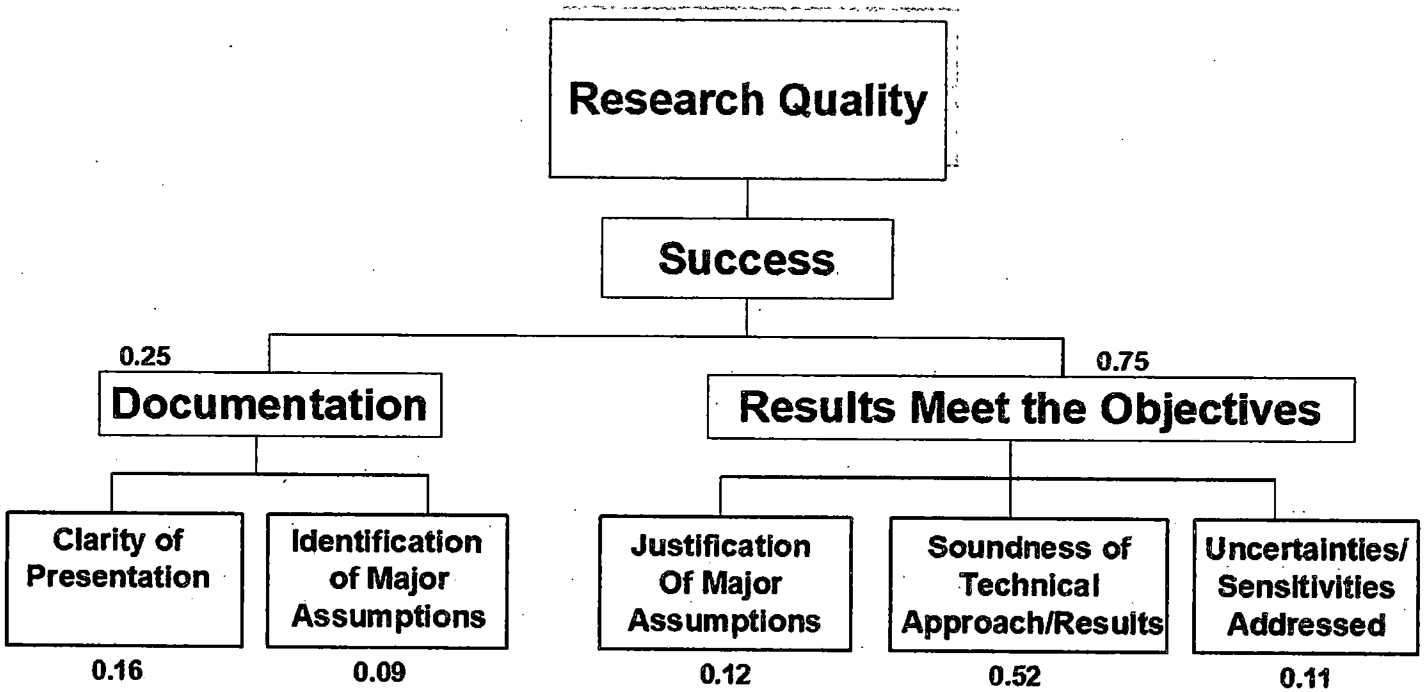
Quality Review Motivation

- Independent evaluation of quality and utility of research projects
- Conforms with Government Performance and Results Act (GPRA)

Quality Review Process

- ACRS typically selects two or three projects proposed by RES
- Three members assigned to each project to complete review
  - RES and sponsoring program office meeting
  - Present report to Full Committee
- Quality rating finalized by Full Committee

Evaluation Criteria and Scoring



- Evaluation emphasizes results meeting project objectives
- Scoring encourages improvement (e.g., "5" is satisfactory)

Quality Review Status

- 2018 review completed
  - NUREG-2218, "An International PIRT Expert Elicitation Exercise for HEAFs"
  - NUREG/CR-7237, "Correlation of Seismic Performance in Similar SSCs"
- 2019 review underway
- Alternate 2020 activity under consideration to provide more strategic input

Research Biennial Review

- 2020 Biennial Research Review underway
- Review continues to emphasize 1997 Commission direction
  - Need, scope, and balance of reactor safety research program
  - Progress of ongoing activities
  - How well RES anticipates research needs and is positioned for changing environment

Research Biennial Review (cont'd)

- Updated 2018 process provides succinct report, also emphasizing:
  - Prioritization and identification of user needs
  - Long-term planning
- Letter report to be issued in March 2020

Thank You!

Acronyms

- ACRS – Advisory Committee on Reactor Safeguards
- AOO – Anticipated Operational Occurrences
- BDBE – Beyond Design Basis Events
- DBE – Design Basis Event
- DCA – Design Certification Application
- DG – Draft Guide
- DHRS – Decay Heat Removal System
- DI&C – Digital Instrumentation and Control
- DID – Defense in Depth
- EA – Exclusion Area
- ECCS – Emergency Core Cooling System
- EDO – Executive Director for Operations
- GPRA – Government Performance and Results Act
- HEAF – High Energy Arc Fault
- LBE – Licensing Basis Event
- LMP – Licensing Modernization Project
- LPZ – Low Population Zone
- MWe – Megawatt (electric)
- MWt – Megawatt (thermal)
- NEI – Nuclear Energy Institute
- NPM – NuScale Power Module
- NPUF – Non-production and Utilization Facility
- NRC – U.S. Nuclear Regulatory Commission
- PCD – Population Center Distance
- PIRT – Phenomenon Identification and Ranking Table
- PRA – Probabilistic Risk Assessment
- PWR – Pressurized Water Reactor
- RES – Office of Nuclear Regulatory Research
- RG – Regulatory Guide
- SER – Safety Evaluation Report
- SMR – Small Modular Reactor
- SSC – Structure, System, or Component
- ST – Source Term
- TVA – Tennessee Valley Authority

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

November 27, 2019

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

SUBJECT: Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99

Dear Ms. Doane:

During the 668th meeting of the Advisory Committee on Reactor Safeguards, November 6-8, 2019, we completed our review of the staff's technical letter report, TLR-RES/DE/CIB-2019-2, "Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide (RG) 1.99." Our Metallurgy & Reactor Fuels Subcommittee reviewed this technical letter report on August 22, 2019. During these meetings, we had the benefit of discussions with the staff and the Electric Power Research Institute (EPRI). We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATION

1. 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).

2. 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.

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

4. 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 reactor pressure vessel (RPV) limits are not challenged.

BACKGROUND

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 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.

DISCUSSION

Regulatory Guide 1.99, Revision 2 ETC Deficiencies

The technical letter report identifies several deficiencies with the current Revision 2. These are:

- Non-conservativism at high fluence for base metals
- Inaccuracies for reactor vessel materials with low copper content
- Underestimated standard deviation relative to the current database
- Conservative bias at low-to-mid fluences
- Lack of temperature adjustment

These deficiencies raise potential safety margin issues because resulting estimates of reference nil-ductility transition temperature shift (ΔRT_{NDT}) may be non-conservative for plants with vessels exposed to higher levels of neutron fluence anticipated during subsequent license renewal. Figure 1 contains plots of the residual of ΔRT_{NDT} for both welds and base metal. Residuals are computed as the difference between the RG 1.99, Revision 2 predicted value and the measured value from the current large embrittlement data base, incorporating both US and international data. A negative residual value indicates non-conservatism. The technical letter report analysis suggests that between fluences of 3×10^{19} n/cm^2 and 6×10^{19} n/cm^2 (E > 1 MeV), the mean residual becomes increasingly non-conservative.

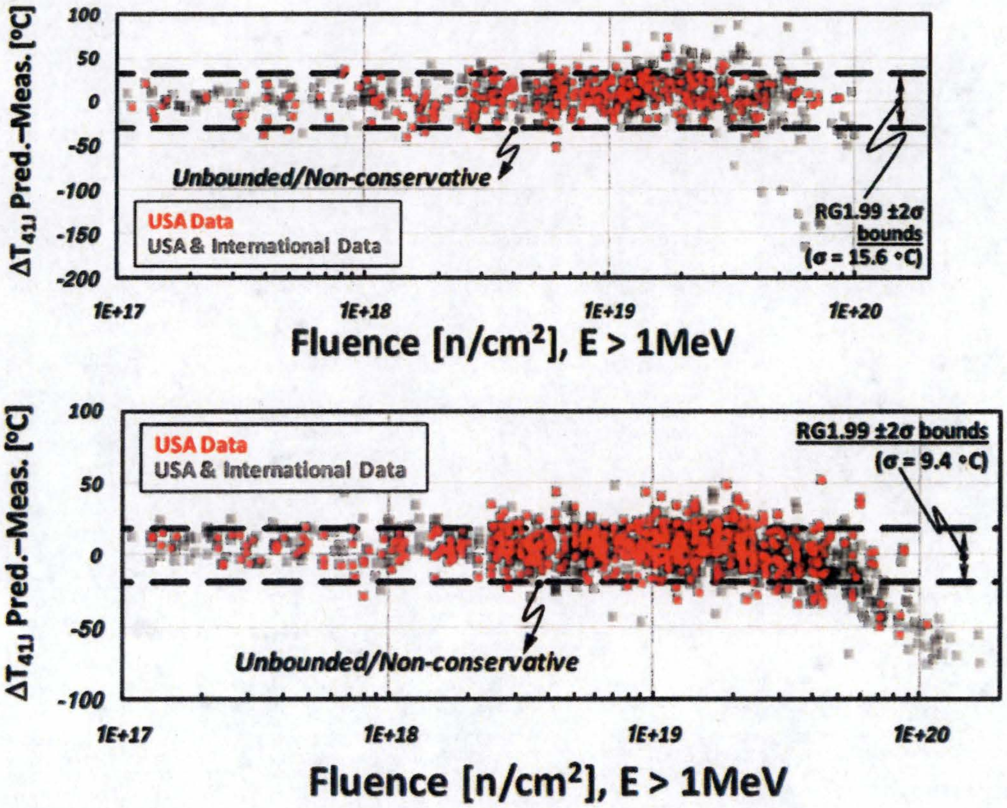


Figure 1. Estimates of the residual (RG 1.99, Revision 2 predicted minus measured values) of ΔRT_{NDT} using the current US and international database for (a) welds and (b) base metal.

Standard deviation values from the RG are used as a defined margin in RPV embrittlement calculations (horizontal dash lines in Figure 1). The current large embrittlement database has a larger standard deviation (greater scatter) than the original database used to develop the ETC in the RG. This represents another potential non-conservativism.

Evaluation of Modern Embrittlement Trend Correlations

The ASTM Subcommittee E10.02, Behavior and Use of Nuclear Structural Materials, has performed an extensive review and comparison of several ETCs, including that in the RG, and the correlation used for the alternate PTS rule (10 CFR 50.61a). The ASTM E900-15,"Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," correlation was found to represent the best fit to the current expanded data base and is more predictive of the behavior at the higher fluences expected during subsequent license renewal. The ASTM E900-15 ETC also has a built-in temperature adjustment term, addressing another of the deficiencies identified in the RG ETC.

Impact on Operating Plants

Electric Power Research Institute (EPRI) estimated the effects of adopting the ASTM E900-15 correlation for RPVs with high or low copper content. In the case of boiling water reactors (BWRs), where the fluence to the reactor vessel is an order of magnitude lower than PWRs, use of the ASTM E900-15 correlation will not be necessary. For PWRs, the effect can

be significant in some cases. Additionally, the limiting RPV component (weld or plate) may change. As projected fluences increase from 10^{19} to 10^{20} n/cm^2 (E > 1 MeV), EPRI results indicate increases in ΔRT_{NDT} of 25° to 75°F above that predicted using the RG (irrespective of the copper content). The NRC staff and EPRI estimate that none of the existing PWR fleet will be affected until circa 2025.

Working and Oversight Groups

The NRC staff has established a working group to complete the following tasks:

- Recommend an alternative ETC.
- Determine limitations of ETC implementation.
- Determine how to apply individual plant surveillance data.
- Determine margins on ETC.
- Determine default values for inputs that are not available.
- Write a draft revised RG for internal review.

The staff working group has adopted the correlation in ASTM E900-15. Completion of the tasks identified by the working group will provide a sound basis for revision of the RG. The working group also presented a schedule for completion of the identified tasks and development of a draft revised guide.

The oversight group is charged with overall supervision of the effort and to provide guidance on implementation. Given that each plant has a different operating temperature, material chemistry, and projected end-of-life fluence, the oversight group should develop an implementation path forward considering each plant's situation. This would eliminate unnecessary burden on plants for which RPV limits are not challenged.

CONCLUSIONS

The ETC in RG 1.99, Revision 2 has a number of deficiencies, the most significant of which is increasing error at high fluence. The correlation in ASTM E900-15, that is based on a much more extensive database, overcomes the deficiencies in this RG and provides the best fit at higher fluences.

A staff working group has been established and has identified a path forward for addressing this issue. A staff oversight group has also been established to guide the implementation of a revised RG.

We look forward to reviewing the updated RG and implementation plan.

Sincerely,

/RA/

Peter Riccardella
Chairman

REFERENCES

1. Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99 Technical Letter Report, TLR-RES/DE/CIB-2019-2, July 31, 2019 (ML19203A089)

2. Assessment of Predictions of RTNDT and Upper Shelf Energy made using Branch Technical Position 5-3, March 23, 2017 (ML16341B108)

3. Regulatory Guide 1.99 Revision 2, Radiation Embrittlement of Reactor Vessel Materials, May 31, 1988 (ML003740284)

4. SECY-09-0059, Final Rule to Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61a) (RIN 3150-A101) April 9, 2009 (ML083470915)

5. NUREG-2163, Technical Basis for Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule – Final Report, September 30, 2018 (ML18255A118)

6. BWRVIP-86, Revision 1-A: BWR Vessel and Internals Project, "Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," Final Report, May 2013 (ML13176A097)

7. ASTM E900-15, Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, February 1, 2015

8. ADJE090015-EA, Technical Basis for the Equation used to Predict Radiation Induced Transition Temperature Shift in Reactor Vessel Materials

9. 10 CFR Appendix G to Part 50 – Fracture Toughness Requirements

November 27, 2019

SUBJECT: Assessment of the Continued Adequacy of Revision 2 of Regulatory Guide 1.99

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NAME	CBrown	CBrown	LBurkhart *(KHoward for)*	SMoore	PRiccardella
DATE	11/25/2019	11/25/2019	11/25/2019	11/25/2019	11/27/2019

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

November 26, 2019

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

SUBJECT: 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"

Dear Ms. Doane:

During the 668th meeting of the Advisory Committee on Reactor Safeguards, November 6-8, 2019, we reviewed the staff's knowledge management publication, 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." Our review was also informed by staff presentations on April 18, 2019, and August 21, 2019, to the Thermal-Hydraulic Subcommittee, during which a draft of the assessment framework was successfully applied to the staff's review of the D5 correlation for SVEA-96 Optima3 fuel.

CONCLUSION AND RECOMMENDATION

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

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

DISCUSSION

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.

Even though NUREG/KM-0013 focuses exclusively on the assessment of critical boiling transition models, this approach could be used for other data-driven models and its use should be explored. For example, a similar approach could be used to assess correlations for experimental material property data, CRUD (Chalk River Unidentified Deposit or Corrosion Residual Unidentified Deposit) deposition and removal from fuel, or thermal-hydraulic parameters such as heat transfer coefficients.

SUMMARY

The credibility assessment framework documented in NUREG/KM-0013 is an innovative technical approach to review the adequacy of data-driven models. Extending this framework to other data-driven model applications should be explored.

Sincerely,

/RA/

Peter Riccardella
Chairman

REFERENCE

1. 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," March 2019 (ADAMS Accession Number ML19073A249).

November 26, 2019

SUBJECT: 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"

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DATE	11/26/2019	11/26/2019	11/26/2019	11/26/2019	11/26/2019

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

November 26, 2019

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

SUBJECT: 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

Dear Ms. Doane:

During the 668th meeting of the Advisory Committee on Reactor Safeguards, November 6-8, 2019, we met with representatives of Mitsubishi Heavy Industries, Ltd. (MHI), and the NRC staff to review the safety evaluation reports (SERs) with no open items associated with the following US-APWR design certification application topics:

- Design Control Document (DCD), Chapter 8, "Electric Power,"
- DCD, Chapter 18, "Human Factors Engineering," and
- Topical Report MUAP-07001, "The Advanced Accumulator."

Our US-APWR Subcommittee reviewed these chapters and the topical report on September 19, 2019. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATION

1. Our 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. We will continue to consider integral effects of system interactions as we complete our final review.

2. Our review of the SER for the topical report on the advanced accumulator did not identify any safety issues.

3. The SERs should be issued.

BACKGROUND

MHI submitted a design certification application for the US-APWR on December 31, 2007. 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 will aid in resolution of concerns and facilitate timely completion of the design certification application review. The staff's SERs and our review of these chapters address Design Control Document (DCD), Revision 4, and supplemental material, including MHI responses to staff requests for additional information.

DISCUSSION

We have previously reviewed these chapters, the topical report and the staff SERs with open items during meetings on October 24, 2008; November 9, 2010; September 8, 2011; September 20, 2012; September 17, 2013; December 5, 2013; and August 20, 2015. The following discussion reflects our reviews of the updated SERs.

Electric Power

Safety-related emergency alternating current (AC) power for the US-APWR is supplied by four gas turbine generators (GTGs). The design includes two additional smaller non-safety GTGs that can be started and aligned manually as alternate AC (AAC) power supplies if the offsite power fails, and power is not available from the safety-related GTGs. The safety-related GTGs are designed to start and be ready to accept load within 100 seconds after loss of power at their respective buses, which is longer than that for a comparably rated diesel generator. The US-APWR design incorporates advanced accumulators that provide extended passive coolant injection, thus allowing a longer interval for GTG starting and loading for all design basis loss of coolant accident (LOCA) conditions.

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.

Human Factors Engineering

Human Factors Engineering (HFE) involves twelve areas of review that are needed for successful integration of human characteristics and capabilities into nuclear power plant design. These areas of review are: HFE Program Management, Operating Experience Review, Functional Requirements Analysis and Function Allocation, Task Analysis, Staffing and Qualifications, Human Reliability Analysis, Procedure Development, Training Program Development, Human-System Interface Design, Human Factors Verification and Validation, Design Implementation, and Human Performance Monitoring.

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.

Advanced Accumulator

In the US-APWR design, the emergency core cooling system (ECCS) injection functions are provided by the high-head safety injection system (HHSIS) and the advanced accumulator (ACC) system. The HHSIS contains four divisions of pumps, normally aligned to deliver water from the refueling water storage pit to the reactor vessel. Four ACCs, one for each loop, passively provide the functions of both conventional accumulators and a low-head safety injection system. Thus, the US-APWR ECCS does not contain a separate low head safety injection system typically found in conventional pressurized water reactors. The ACCs are designed to provide injection flow at a high rate to rapidly refill the reactor vessel lower plenum and the downcomer during the blowdown phase of a large-break LOCA. This initial large injection flow is followed by passive switching to a much lower injection flow rate needed to maintain downcomer level during the core reflood phase. The accumulator is designed to ensure that the calculated peak cladding temperature and cladding oxidation level remain within acceptable regulatory limits in both the blowdown and reflood phases.

MHI initially qualified the ACC using a combination of testing and computational fluid dynamics (CFD) analysis. We reviewed the previous qualification analysis that used half-scale testing with extrapolation to full scale using CFD. Although the methodology was acceptable, we concurred with the staff's recommendations to increase the uncertainties that are used in 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.

SUMMARY

Our review of the SERs for the topical report on advanced accumulators and Chapters 8 and 18 did not identify any safety issues that would preclude issuance of a design certification at this stage. The staff safety evaluation reports should be issued.

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

Sincerely,

/RA/

Peter Riccardella
Chairman

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1. Mitsubishi Heavy Industries, LTD., Design Control Document for the US-APWR, September 19, 2013 (ML13262A488)

2. U.S Nuclear Regulatory Commission, US-APWR Design Certification Application – Safety Evaluation with no Open Items for US-APWR Design Certification Safety Evaluation Chapter 8, "Electric Power," August 18, 2019 (ML19219A217)

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4. U.S Nuclear Regulatory Commission, Advanced Pressurized Water Reactor Advanced Topical Report Safety Evaluation for Topical Report MUAP-07001, Revision 7, "The Advanced Accumulator," July 29, 2019 (ML19154A122)

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November 26, 2019

SUBJECT: 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

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

November 26, 2019

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

SUBJECT: 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

Dear Ms. Doane:

During the 668th meeting of the Advisory Committee on Reactor Safeguards (ACRS), November 6-8, 2019, we completed our review of the staff's safety evaluation (SE) of Brunswick Steam Electric Plant (BSEP) Units 1 and 2 license amendment request (LAR) to allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM™11 fuel. Our Thermal-Hydraulic Subcommittee also reviewed this LAR on November 5, 2019. As part of our review, we met with the Nuclear Regulatory Commission (NRC) staff and representatives from Framatome and Duke Energy. We also had the benefit of the referenced documents.

CONCLUSION AND RECOMMENDATION

1. 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.

2. The LAR should be approved, and the SE should be issued.

BACKGROUND

BSEP Units 1 and 2 are of the BWR/4 design with Mark I containments. These units began commercial operation with a thermal power of 2436 MWt in 1975 (Unit 2) and in 1976 (Unit 1). In May 2002, NRC granted these units an extended power uprate (EPU) to increase to the current licensed thermal power of 2923 MWt, and in September 2018, NRC approved BSEP to operate in the Maximum Extended Load Line Limit Analysis Plus domain with Framatome ATRIUM™ 10XM fuel. Currently, BSEP uses a mixture of Framatome and General Electric Hitachi methods to demonstrate that safety margins are maintained for each new core reload. In October 2018, Duke Energy submitted an LAR to adopt advanced Framatome methods for

fuel and thermal-hydraulic performance to the latest generation ATRIUM™ 11 fuel. Duke Power indicated that they plan to start loading this new fuel type during the 2020 BSEP outage.

DISCUSSION

This LAR is the first application of eight new analysis methodologies. Of these eight, five were approved on a generic basis at the time of submittal:

- AURORA-B methodology for anticipated operational occurrences (AOO) and anticipated transients without scram (ATWS) over-pressure events.
- Control rod drop accident (CRDA) method.
- AREVA-approved methods with incorporation of Chromia-doped fuel properties.
- Realistic thermal-mechanical fuel rod methodology.
- ACE/ATRIUM™ 11 critical power correlation method.

Two of the eight methodologies were not approved when this BSEP LAR was submitted, but have since been approved:

- AURORA-B loss of coolant accident (LOCA).
- ATWS with instability (ATWS-I).

One methodology is currently under review by the staff for generic use, and the applicant requested that it be approved for use in BSEP as part of this LAR:

- Best-estimate Enhanced Option III with Confirmation Density Algorithm (BEO-III/CDA) stability solution.

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; AOOs; ATWS 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.

Generic SEs of the ATWS-I and LOCA methods had not been completed at the time of LAR submittal; therefore, the staff conducted a detailed review to confirm that their use for BSEP was acceptable. In the case of ATWS-I methods, the staff also performed confirmatory calculations to verify that these methods yielded acceptable results. Recently, we have completed reviews of the generic methods for LOCA, AOO, and ATWS-I and concurred with the staff that these generic methods should be approved. The BSEP-specific version of these methods in the LAR is essentially identical to the generic version that we reviewed and should also be approved.

The staff review of the generic BEO-III stability methodology is currently underway, and this section of the LAR was approved on a plant-specific basis with six licensing conditions. BEO-III is implemented by performing calculations at a sufficient number of cycle exposure points. The

goal of BEO-III is to determine the value of the operating limit minimum critical power ratio (OLMCPR) that guarantees that, even if an instability occurs, the safety limit is not challenged. The methodology calculates a best-estimate OLMCPR plus its uncertainty using non-parametric statistical analyses. The typical reload application will perform many separate calculations, which should cover any expected instability scenario. The staff has reviewed the BSEP-specific implementation of BEO-III and found it acceptable.

Framatome provided an example application for Cycle 23 of BSEP Unit 1 showing that the BEO-III OLMCPR value is not limiting. Other AOOs (e.g., load rejection or feedwater controller failure) require larger OLMCPR values to ensure the safety limit is not challenged.

The staff reviewed the Reload Safety Analysis Report (RSAR) for Cycle 23 of BSEP Unit 1. The cycle-specific results in the RSAR confirmed all limits were met for the full range of operating conditions.

SUMMARY

The Framatome core reload analysis methodology is acceptable for use in BSEP Units 1 and 2 licensing applications that incorporate ATRIUM™11 fuel. The LAR should be approved, and the SE should be issued.

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

Sincerely,

/RA/

Peter Riccardella
Chairman

REFERENCES

1. U.S. Nuclear Regulatory Commission (NRC), "Draft 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," October 29, 2019 (ADAMS Accession No. ML19255H826).
2. U.S. NRC, Advisory Committee on Reactor Safeguards, "Safety Evaluation for Topical Report ANP-10300P, Revision 0, 'Aurora-B: an Evaluation Model for Boiling Water Reactors; Application to Transient And Accident Scenarios,'" October 19, 2017 (ADAMS Accession No. ML17290B212).
3. U.S. NRC, Advisory Committee on Reactor Safeguards, "Safety Evaluation for ANP-10333P, Revision 0, & Aurora-B: an Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," March 26, 2018 (ADAMS Accession No. ML18081A291).
4. U.S. NRC, Advisory Committee on Reactor Safeguards, "Safety Evaluation for ANP-10332P, Revision 0, Aurora-B: An Evaluation Model for Boiling Water Reactors; Application to Loss-of-Coolant Accident Scenarios," February 22,2019 (ADAMS Accession No. ML19057A018).

5. U.S. NRC, Advisory Committee on Reactor Safeguards, "Safety Evaluation of Topical Report ANP-10346P, Revision 0, 'ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA,'" November 4, 2019 (ADAMS Accession No. ML19308A004).

6. Duke Energy, "Brunswick, Units 1 and 2, Request for License Amendment Regarding Application of Advanced Framatome Methodologies," October 11, 2018 (ADAMS Accession No. ML18284A394).

7. Duke Energy, "Brunswick, Units 1 and 2 - Update to Request for License Amendment Regarding Application of Advanced Framatome Methodologies," November 28, 2018 (ADAMS Accession No. ML18333A028 Proprietary).

8. Duke Energy, ANP-3772P, Revision 0, "CR Supplement Report on Brunswick LAR Analyses," May 14, 2019 (ADAMS Accession No. ML19135A030 Proprietary).

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13. Framatome Inc., ANP-2637P, Revision 7, "Boiling Water Reactor Licensing Methodology Compendium," September 30, 2018 (ADAMS Accession No. ML18264A016 Proprietary).

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15. Duke Energy, "Brunswick Steam Electric Plant, Units 1 and 2, Updated Final Safety Analysis Report, Revision 26," August 13, 2008 (ADAMS Accession No. ML18249A165).

16. Framatome Inc., ANP-3702P Revision 0, "Brunswick ATRIUM 11 Transient Demonstration", August 2018 (ADAMS Accession No. ML18284A394, Attachment 12.a, Proprietary).

17. Framatome Inc., ANP-3703P Revision 0, "Best Estimate Option-III Analysis Methodology for Brunswick Using RAMONA5-FA," August 2018 (ADAMS Accession No. ML18284A394, Attachment 15.a, Proprietary).

18. Framatome Inc., ANP-3705P Revision 0, "Applicability of Framatome BWR Methods to Brunswick with ATRIUM 11 Fuel," September 2018 (ADAMS Accession No. ML18284A394, Attachment 5.a, Proprietary).

19. Framatome Inc., ANP-3674P Revision 1, "Brunswick Units 1 and 2 LOCA Analysis for ATRIUM 11 Fuel," October 2018 (ADAMS Accession No. ML18284A394, Attachment 13.a Proprietary).

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November 26, 2019

SUBJECT: 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

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

November 4, 2019

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

SUBJECT: REVIEW OF ADVANCED REACTOR COMPUTER CODE EVALUATIONS

Dear Chairman Svinicki:

During the 667th meeting of the Advisory Committee on Reactor Safeguards, October 2-4, 2019, we reviewed the staff's evaluations of computer codes to be used for analyses of advanced non-light water reactors (non-LWRs). Our Future Plant Designs Subcommittee also reviewed this matter during meetings on May 1 and September 17, 2019. Previously we were briefed by the U.S. Department of Energy (DOE) concerning the capabilities of their computer codes on August 21 and November 16, 2018. During these meetings we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATIONS

The four draft documents on Strategy 2 of the Vision and Strategy document provide an evolving exposition of the staff's approach for the identification and assessment of non-LWR computer codes and data that may be used to support licensing reviews of non-LWR submittals.

1. The approach taken by the staff supports their readiness to review submittals for non-LWR designs of many different types. This approach can also help the staff understand the new reactor designs and associated phenomena.

2. Ideally, the tools for staff confirmatory analysis should be as independent as practical, validated, understood by the staff, and usable on the staff's computer resources.

3. The staff also needs to become sufficiently familiar with applicant codes to support timely reviews of submitted analyses.

4. The overview report should be revised to better explain how this approach integrates the evaluations discussed in Volumes 1-3 and to present a coherent strategy for evaluations. Four principles should underlie the strategy: simplicity, completeness, working the problem backwards starting with the source term, and scaling down the level of effort of licensing review proportionately as the hazard decreases.

5. The staff should perform pilot studies using relatively mature designs to illustrate how the analysis should proceed. They should consider a case using the licensing modernization project (LMP) and one that uses an alternative approach. This would increase confidence in the overall approach being pursued by staff and flush out any needed refinements.

BACKGROUND

The NRC staff developed a report in 2016, "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Mission Readiness," as the staff contemplated how to review and regulate a new generation of non-LWRs including their associated fuel cycles and waste forms. The report lays out six strategies to accomplish its goals and provides a set of Implementation Action Plans that identify specific, actionable tasks that can fulfill the strategies. Over the intervening years, we have provided letter reports on the vision and strategy document as well as several products of the Implementation Action Plans—non-LWR design criteria, functional containment performance criteria, the licensing modernization project (LMP, now called the "Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,") and siting for advanced reactors.

The subject of our current review is a series of draft documents prepared by the staff on evaluating computer codes needed to conduct confirmatory analyses of non-LWR nuclear power plants. The draft documents reviewed are entitled, "Code Assessment Plans for NRC's Regulatory Oversight of Non-Light Water Reactors," "NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 1 - Computer Code Suite for Non-LWR Design Basis Event Analysis," "NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 2 – Fuel Performance Analysis for Non-LWRs," and "NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 - Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis."

The draft documents were prepared under Strategy 2 of the Vision and Strategy document. The main goal of Strategy 2 is to identify and develop the tools and databases that will optimize regulatory readiness and assist the NRC staff in performing its safety reviews of non-LWR license applications. Central to Strategy 2 are the selection and development of computer codes to be used for confirmatory analyses of non-LWR designs.

DISCUSSION

The staff has completed a major step in addressing Strategy 2: Computer Codes. The four draft documents they shared with us complete a major portion of their near-term action plan by identifying and assessing the available computer codes and databases. This activity is helping the staff develop their understanding of the technologies involved and the associated phenomena that will be encountered in reviewing non-LWR designs. Two final documents, Volume 4 on licensing and siting dose assessment codes and Volume 5 on fuel cycle topics, have not been completed. The gap analyses associated with Volumes 1-3 consider both knowledge gaps (fundamental physics and chemistry) and computer code gaps. Staff stated that most of the knowledge gaps are in the area of severe accidents. They have been comprehensive in identifying potential gaps, consistent with Phenomena Identification and Ranking Tables (PIRTs) that have been performed in recent years. Knowledge gaps will need to be addressed by experiment and operating history, perhaps as interpreted by expert elicitation. Computer code gaps have been addressed in the current drafts. The staff's contractor developed a predictive capability maturity model (PCMM) to characterize the state of

readiness of the computer codes. It generates maturity level scores over a set of six fundamental modeling and simulation elements for each reactor type. It has been useful for evaluating the level of effort expected to complete development of the computer codes for use in staff reviews.

The staff evaluated computer codes from four sources—NRC codes, DOE advanced multi-physics codes, commercial codes, and international codes. During the period of our review, the staff was rapidly developing its evaluation, drafting the four reports, and conducting preliminary testing. As part of their approach, the staff has assembled a suite of primarily NRC and DOE codes they call BlueCRAB as a comprehensive reactor analysis bundle. The set includes a commercial computational fluid dynamics (CFD) code and one international code for nuclear cross-sections, as well as NRC and DOE codes for cross-sections, system and core thermal hydraulics, CFD, neutronics, thermal hydraulics and fuel performance. Almost all have now been coupled primarily through the DOE MOOSE (multiphysics object-oriented simulation environment) platform, a fully coupled, fully implicit solver that allows independent codes to be coupled and exchange information. BlueCRAB includes no codes for source term evaluation; the staff will rely on their own code MELCOR.

The primary goal of Strategy 2, and indeed of the entire vision and strategy process, is readiness. The staff needs to be ready to review non-LWR submittals and to perform confirmatory analyses. This implies that they must have an in-depth understanding of how each design works. The staff sees the need to be prepared for a range of review strategies and regulatory concerns. The specific approach will depend on applicant submittals.

Staff considers its readiness highest for high-temperature, gas-cooled reactors and sodium fast reactors. For gas-cooled fast reactors, heat pipe based microreactors, molten salt reactors, and molten salt-fueled reactors substantial development will be required. For the more exotic of these technologies, the staff anticipates substantial savings if DOE codes are selected for confirmatory analysis. Some of our members are not so sanguine.

The staff readiness goals suggest that all designs should be considered equally likely for submittal. However, it is clear that such an approach is neither practical nor possible. The time to make necessary improvements to the codes and ensure staff competence in their use cannot be reconciled with such an egalitarian point of view. Near-term submittals are expected, and the staff must set priorities based on best judgment. The staff is holding discussions with DOE to identify data gaps and set high-priority needs for obtaining such data.

Because this work is of high priority and time is believed to be short, many tasks are being performed in parallel. In the rush to complete, it is not surprising to find some inconsistencies.

The technical volumes of the draft report identify key issues of readiness for three types of analysis and the associated computer codes:

- Volume 1 considers codes for non-LWR design basis event analysis—systems analysis—evaluating how each machine works, whether the safety functions and systems are acceptable, and if the operating limits are met. Volume 1 presents the comparative maturity evaluations for each reactor type but draws no conclusions about which codes should be used. The variety of detailed computational tools in BlueCRAB may be needed to verify that some advanced reactor safety functional and operational limits are met.

- Volume 2 considers fuel performance codes for non-LWRs by performing a comparison of current fuel performance codes, NRC's FAST and DOE's BISON. They have chosen FAST for their confirmatory analysis tool for fuel performance. They used the PCMM model this time to plan necessary development activities throughout the next two and a half years.

- Volume 3 considers non-LWR code development for severe accident progression, source term, and consequence analysis, evaluating what is the fission product inventory, its transport, and the resulting source term. Volume 3 lays out the regulatory needs, the required development activities for each reactor type, and a development plan for that work over the next two and a half years. Staff identified their systems analysis code, MELCOR, that can perform reactor severe accident progression and source term analysis.

Early on, we had a sense that the staff might be force-fitting some of the detailed DOE codes to meet their need for confirmatory analysis. Our members offered many comments suggesting that we may not need all the detail in the DOE codes for deciding safety issues. The staff should decide where the current codes are good enough for safety findings and where the new codes could become necessary or advantageous. Over the four months between our subcommittee meetings, the staff's view of the best path forward appeared to evolve and coalesce. Staff agrees that DOE codes provide exceptional detail and acknowledge that such details might be very essential for a core designer who wants to minimize the number of feed assemblies or to optimize performance. However, as they are evaluating margins to various safety limits, they generally will not need that detail.

The staff considers that they need two sets of tools—one for adequacy of safety functions and operational limits and another for source term. Ideally, the tools for staff confirmatory analysis should be as independent as practical, validated, understood by the staff, and usable on the staff's computer resources. The staff also need to become sufficiently familiar with applicant codes to support timely reviews of submitted analyses.

We and the staff concur that expanding the overview report to clarify the staff's strategy on selecting computer codes for confirmatory analysis would be helpful, as would explaining why different criteria might apply for different applications. Four principles suggest the way to judiciously move forward: simplicity, completeness, working the problem backwards, and a graded, risk-informed licensing review.

The philosophy should be to start simple and only get detailed as needed. If the margins for the new reactors are substantial, simple analyses can confirm that safety is maintained. With some of the new designs and an expected lack of data, a judicious consideration of uncertainty in the associated calculations will be important. An example of an area that affects the complexity of analysis and code use is the issue of design basis accidents (DBAs) and beyond design basis accidents (BDBAs). The continued separation of DBA and BDBA codes may not be necessary for some non-LWRs. Modeling of accident phenomena that occur in LWRs changes significantly from DBA to BDBA regime because of rod ballooning, loss of geometry, oxidation, and meltdown. Some of the advanced reactors may have a much more graceful change in accident space and this should allow one toolset to do all the calculations. This characteristic should enable a great simplification of the analysis.

Completeness is essential to ensure we have identified the most risk-significant scenarios. The process described in the LMP provides a systematic way to identify initiating events and scenarios that can lead to release. Any alternative must use a systematic search and not rely on past experience with LWRs.

The best way to approach the stepwise application of increasing detail could be to work the problem backwards. By that we mean start with the source term. Depending on its associated hazard and the ability to mobilize it, a simple bounding analysis could show that safety criteria are met. If not, then stepwise increase the detail and realism of the analysis.

The final principle relates to scaling down the level of effort of licensing review proportionately as the hazard decreases. The staff should find a way to make the licensing effort commensurate with the associated risk. Something akin to the approach used for research reactors could be considered, which would greatly simplify the analyses.

Finally, we recommend pilot studies to illustrate how the analysis should proceed using relatively mature designs. Such studies can provide insights about the required level of detail, the importance of modeling parameters, and prioritization of data needs. These pilots are an important step for increasing confidence in the overall approach being pursued by staff for identifying any needed refinements. The staff should consider a case using LMP and one that uses an alternative approach.

SUMMARY

The staff has made significant progress in the challenging area of Strategy 2 for developing staff readiness to review non-LWR submittals. However, there is much more to accomplish in a relatively short time. We look forward to continued interactions with the staff.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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November 4, 2019

SUBJECT: REVIEW OF ADVANCED REACTOR COMPUTER CODE EVALUATIONS.

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

November 04, 2019

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

SUBJECT: SAFETY EVALUATION OF TOPICAL REPORT ANP-10346P, REVISION 0, "ATWS-I ANALYSIS METHODOLOGY FOR BWRs USING RAMONA5-FA"

Dear Ms. Doane:

During the 667th meeting of the Advisory Committee on Reactor Safeguards, October 2-4, 2019, we reviewed the staff's safety evaluation report of Framatome topical report ANP-10346P, Revision 0, "ATWS-I Analysis Methodology for BWRs using RAMONA5-FA." Our Thermal Hydraulic Subcommittee also reviewed this topical report on August 21, 2019. During these meetings, we had the benefit of discussions with the staff and representatives from Framatome. We also had the benefit of the referenced documents.

Conclusion and Recommendation

1. The RAMONA5-FA methodology to analyze anticipated transients without scram with instability (ATWS-I), when used in compliance with the seven limitations and conditions imposed by the staff, is acceptable for use in boiling water reactor (BWR) licensing applications.

2. The safety evaluation should be issued.

Background

RAMONA is a family of codes. In the U.S., RAMONA-III was first acquired by the U.S. NRC from Scandpower, Norway, in 1979. Based on this code, Brookhaven National Laboratory (BNL) developed RAMONA-3B/MOD0. BNL later incorporated three-dimensional (3D) neutron kinetics and other model improvements to develop RAMONA-4B. RAMONA5-FA is a Framatome proprietary version, which has been approved for use in some licensing applications, including calculation of setpoints for stability solution implementations. The AISHA and SINANO codes described in ANP-3274P-A were approved for use to analyze ATWS-I events for extended flow window (EFW) applications at Monticello Nuclear Generating Plant, Unit 1.

The AISHA and SINANO models and other improvements have been incorporated in the RAMONA5-FA ATWS-I methodology on a generic (i.e., non-plant specific) basis. The current

topical report, ANP-10346P, documents this update and provides an ATWS-I phenomena identification and ranking table (PIRT), a summary of the validation for the methodology, and a description of the analysis procedure.

Discussion

The staff has previously approved multiple components of the RAMONA5-FA ATWS-I methodology as part of their review of the Monticello EFW license amendment request; therefore, the primary focus of the staff review was the aspects of this methodology that are novel to ensure applicability on a generic basis, as well as the integration of multiple methodologies developed at different times into a single approach for generic ATWS-I analyses. The staff also reviewed the ATWS-I PIRT, experiment benchmarking, and an example plant application. The staff review followed key elements of the evaluation model development and assessment process outlined in Regulatory Guide 1.203, including: accident scenario description and phenomena identification and ranking; evaluation methodology; code assessment; uncertainty analysis; and documentation.

Main features of the RAMONA5-FA ATWS-I methodology include: adaptive 3D nodal diffusion with two-energy groups tightly coupled to the thermal hydraulics solution; automated coupling to MICROBURN-B2 cross sections; random noise models that excite all neutronic modes; nonequilibrium thermal hydraulic models to allow vapor superheat; and a numerical solution that allows for reverse flow and prevents singularities. The control systems and vessel models have been improved to track water level and feedwater temperature to simulate operator mitigation actions more accurately. The fuel thermal-mechanical models have been updated based on the RODEX4 and XEDOR methodologies to include thermal conductivity degradation, gap conductance, and chromium-doped pellet properties. As with all versions of RAMONA, it has the same basic limitation that sacrifices pressure-wave tracking in favor of a more robust solution for momentum conservation.

One of the most significant modifications is the use of the new CPROM critical power ratio correlation to integrate transient dryout and rewet phenomena into a single methodology, without a minimum stable film boiling temperature correlation. CPROM has been developed based on proprietary data from the Karlstein Thermal Hydraulic (KATHY) test facility. The post-dryout heat transfer models are based on KATHY ATRIUM-fuel-specific measurements. The staff has reviewed in detail the models in the RAMONA5-FA ATWS-I methodology and found them acceptable for their intended use.

Framatome has performed code assessment and validation against an extensive set of experimental data, most of them ATRIUM-specific. The data include: void fraction, pressure drop, flow stability tests, and transient dryout-rewet. Benchmarks were also performed against integral tests for a number of plant linear instabilities, and nonlinear plant instability events. Framatome has addressed uncertainties via sensitivity analyses, including sensitivity to nodalization and integration time step. The staff has reviewed the RAMONA5-FA ATWS-I validation and found it acceptable.

The staff has found the RAMONA5-FA methodology acceptable for ATWS-I calculations with seven limitations and conditions: the gap conductance sensitivity shall be reevaluated for new fuels; justification must be provided to demonstrate adequate margin in operator action timing; the assumptions employed in the analysis of record must be verified for core specific applications; transition cores must have additional verification; both turbine trip and recirculation

pump trip must be analyzed to determine the limiting ATWS-I event; plant-specific steam line and valve models must be verified; and plant-specific applications must justify the selected settings for RAMONA5-FA. We concur with these limitations and conditions.

Summary

The RAMONA5-FA methodology to analyze anticipated transients without scram with instability, when used in compliance with the seven limitations and conditions imposed by the staff, is acceptable for use in BWR licensing applications. The safety evaluation should be issued.

Sincerely,

/RA/

Peter Riccardella
Chairman

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November 4, 2019

SUBJECT: SAFETY EVALUATION OF TOPICAL REPORT ANP-10346P, REVISION 0, "ATWS-I ANALYSIS METHODOLOGY FOR BWRs USING RAMONA5-FA"

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

October 31, 2019

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

SUBJECT: ADVANCED BOILING WATER REACTOR DESIGN CERTIFICATION RENEWAL

Dear Chairman Svinicki:

During the 667th meeting of the Advisory Committee on Reactor Safeguards (ACRS), October 2-4, 2019, we completed our review of the design certification renewal application for the advanced boiling water reactor (ABWR) and the associated final safety evaluation report. Our review considered actions by GE-Hitachi (GEH), the first vendor in the U.S. to apply for a design certification renewal. Our ABWR Subcommittee reviewed this matter during a meeting on August 23, 2019. During our review, we had the benefit of discussions with representatives of the staff and GEH. We also had the benefit of the referenced documents.

This report fulfills the requirement of Title 10 of the *Code of Federal Regulations* (10 CFR) 52.57(c) that the ACRS report on those portions of the application which concern safety.

CONCLUSION AND RECOMMENDATION

Staff supplemental safety evaluations (SEs) approved GEH proposed design changes to update and amend specific design attributes that meet the criteria for a Design Certification Renewal in accordance with 10 CFR 52.59, extending it for an additional 15 years, following implementation of the design certification final rule.

1. There is reasonable assurance that the ABWR, under the renewed design certification, can be constructed and operated without undue risk to the health and safety of the public.

2. We concur with the conclusions of the staffs' supplemental renewal SEs to NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," with no open items. The SEs should be issued, and the GEH application for the Design Certification Renewal of the ABWR should be approved.

BACKGROUND

Previously, on July 13, 1994, the U.S. Nuclear Regulatory Commission (NRC) issued the final design approval, along with NUREG-1503, "Final Safety Evaluation Report Related to the

Certification of the Advanced Boiling Water Reactor Design." On May 12, 1997, the NRC issued the final design certification rule for the ABWR design.

On December 7, 2010, GEH requested the NRC to renew the ABWR design certification. The ABWR design certification rule, effective June 11, 1997, would otherwise expire at the end of a period of 15 years, or June 11, 2012. GEH applied for a design certification renewal on December 7, 2010. On July 20, 2012, staff identified proposed changes including Fukushima Near Term Task Force Recommendations. GEH provided the ABWR design control document (DCD), Revision 6, in response to staff requested changes. On June 28, 2019, the staff completed the SEs with no open items.

DISCUSSION

The regulatory basis for renewal of a design certification includes three change categories: modifications, renewal backfits, and amendments. Modifications to the certified design are those changes in accordance with 10 CFR 52.57(a) (e.g., clarifications, changes to correct known errors, typographical errors, or defects that are necessary to meet 10 CFR 52.59(a)). Modifications must comply with the regulations applicable and in effect at the time the certification was originally issued. Renewal backfits are those changes that are necessary to comply with additional requirements imposed by the NRC through application of the criteria in 10 CFR 52.59(b). Amendments are those changes proposed by the design certification renewal applicant in accordance with 10 CFR 52.59(c). Amendments must comply with regulations applicable and in effect at the time of renewal. The GEH Design Certification renewal application contains modifications and amendments but no backfits.

The key significant renewal design changes involved the following areas: amendment to the emergency core cooling system (ECCS) suction strainers; peak cladding temperature (PCT) modification; Fukushima design enhancements; aircraft impact assessment; and containment overpressure protection system (COPS) modification.

- In accordance with guidance of Regulatory Guide 1.82, Revision 4, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," the staff confirmed that the ECCS suction strainer design complied with 10 CFR 50.46(b)(5), providing adequate Net Positive Suction Head margins. The staff also confirmed that GEH addressed the chemical, in-vessel, and ex-vessel downstream effects.
- Following incorporation of the effects of the ECCS evaluation model changes, and correction of errors since the original ABWR design certification, the estimated PCT increased by a small amount (42°C or 75°F). PCT is now 663 °C (1225 °F), which remains in compliance with criteria in 10 CFR 50.46(3)(i).
- To allow combined license applicants to meet anticipated requirements of the Mitigation of Beyond-Design-Basis Events Rule, GEH made design amendments, such as additional non-safety-related water and electrical connections.
- GEH performed a detailed aircraft impact assessment. The staff found that GEH adequately described the key design features and functional capabilities identified and credited to meet 10 CFR 50.150(b), including how the key design features meet the acceptance criteria in 10 CFR 50.150(a)(1).
- GEH modified the COPS design to include a dedicated containment vent path to prevent containment over pressure. The staff concluded that this modification did not alter the safety findings made in NUREG-1503.

In total, 39 design items were reviewed and approved by the staff in supplemental SEs to NUREG-1503 or closed by letter. In addition to reviewing DCD, Revision 6, and responses to requests for additional information, the staff performed audits to resolve outstanding technical issues.

SUMMARY

The staff made safety determinations on the specific modifications and amendments proposed by GEH as part of its design certification renewal application; they were found to meet applicable regulatory requirements. We agree with the staff's determinations. There is reasonable assurance that the ABWR, under the renewed design certification, can be constructed and operated without undue risk to the health and safety of the public.

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

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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October 31, 2019

SUBJECT: ADVANCED BOILING WATER REACTOR DESIGN CERTIFICATION RENEWAL

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

October 17, 2019

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

SUBJECT: ACRS ACTIVITIES TO SUPPORT NRC TRANSFORMATION

Dear Chairman Svinicki,

In response to our recent interactions with the Commission, we engaged in a number of activities to better understand the NRC transformation initiative and assess how the Committee might become more effective and efficient. We were briefed by senior NRC staff on planned and ongoing transformation efforts and conducted a number of ACRS retreats to discuss the topic. We also obtained input from the Executive Director for Operations (EDO), all current Commissioners and several former Commissioners regarding those ACRS activities or products they found to be most effective and impactful in fulfilling our statutory mission. In addition, we had the benefit of the documents referenced.

CONCLUSIONS AND PROPOSED ACRS ACTIONS

1. We will stay abreast of staff transformation initiatives through periodic update meetings and will evaluate how we can continue to contribute to agency transformation activities.

2. We will further improve our effectiveness and efficiency through prioritization of our reviews and independent advice on issues related to public health and safety, emphasizing risk-significant issues and agency transformation priorities.

3. We have identified and begun to implement a number of actions that improve our operational efficiency. Our operating costs have seen significant reductions and continue to trend downwards.

4. Our reviews provide an integrating perspective and increase expectations regarding the quality and rigor of the work performed by both staff and industry. We will continue to provide the depth and breadth necessary to maintain these expectations.

5. We do not see, at this time, a need for modifications or updates to matters referred to the Committee as established in NRC's regulations to implement these actions.

ACRS HISTORICAL PERSPECTIVE

Nuclear energy is unique because of potential hazards related to releases of radioactive material from facilities under normal operations and accidents. Recognizing this, the Atomic

Energy Commission established special advisory committees to advise on the siting of nuclear power plants and review and evaluate the hazards associated with this technology. In 1957, Congress made the Advisory Committee on Reactor Safeguards (ACRS) a statutory committee, composed of experts representing many technical perspectives to provide independent advice.

Currently, the ACRS provides independent advice to the Commission on the safety of proposed or existing NRC licensed facilities and the adequacy of proposed safety standards. The ACRS reviews power reactor and fuel cycle facility license applications for which the NRC is responsible and safety- and risk-significant NRC regulations and guidance relating to these facilities. The ACRS also reports to and advises the Commission on issues associated with nuclear materials and waste management. On its own initiative, ACRS may conduct reviews of specific generic matters or nuclear facility safety-related items.

The Commission may also refer other matters to ACRS for review. The ACRS also provides advice on naval reactor designs. Upon request, ACRS may provide advice on hazards associated with the Department of Energy nuclear activities and facilities, and to the Defense Nuclear Facilities Safety Board.

ACRS operations are governed by the Federal Advisory Committee Act (FACA), which is implemented through NRC regulations (10 CFR Part 7). ACRS operational practices encourage the public, industry, state and local governments, and other stakeholders to express their views on regulatory matters.

NRC TRANSFORMATION INITIATIVES

In January 2018, the EDO tasked a Transformation Team to identify changes to the regulatory framework, culture and infrastructure to further enhance the Agency's effectiveness, efficiency and agility. Techniques, ideas, and information relating to novel technologies from internal and external stakeholders were analyzed to identify specific areas to initiate transformation at the NRC. Initiatives the staff is addressing include:

- Development of an agency-wide process to expand the systematic use of qualitative and quantitative risk and safety insights;
- Allowance of additional flexibility under 10 CFR 50.59, "Changes, Tests, and Experiments," for licensees to make facility changes without prior NRC approval;
- Development of a risk-informed, performance-based approach for licensing of non-light water reactors; and
- Development of regulatory guidance based on application of instrumentation and control (I&C) fundamental design principles.

Staff is currently developing strategies to pursue these and other initiatives with a focus on improving the agency's ability to adapt and transform its culture to that of a modern, risk-informed regulator.

ACRS ROLE IN A TRANSFORMED AGENCY

As the agency develops a vision and strategy to assure that the NRC is ready to review potential applications for advanced, non-light water reactor (non-LWR) technologies, the role of the ACRS, with its diverse technical expertise, to perform integrated/multi-disciplinary reviews continues to be essential.

Commissioner Input

Commissioner input has identified the following areas where ACRS engagement is most effective:

- ***Risk-Informed Decision Making*** – Several commissioners opined that the most important role the ACRS can play is to continue its firm support of and advice regarding risk-informed decision making.
- ***Digital I&C*** – This was identified as an important topic, and one commissioner stated that NRC and the nuclear industry are far behind where they should be on this topic.
- ***Research Reviews*** – Research reviews were identified by several commissioners as an important area for continued ACRS involvement.
- ***New Technologies and Reactor Types*** – ACRS is a highly competent group of dedicated experts from outside the agency, encompassing a broad range of disciplines, who dig deeply into the matters subject to the Committee's review. The ACRS members' independent technical assessments assist the NRC staff in making high-quality regulatory evaluations.

Prioritization of ACRS Review Activities

The following criteria will be used to set priorities for our in-depth reviews:

- Does the issue affect public health and safety?
- Does the issue relate to one of the four agency transformation initiatives (i.e., risk-informed decision making; 10 CFR 50.59 flexibility; licensing of non-LWRs; or digital I&C safety design principles)?
- Does the issue involve new methods or technologies, or is it a routine matter that we have reviewed numerous times before, and for which the staff processes are mature and technically advanced?
- Is the activity directed by the Commission?
- Other criteria that future staff transformation activities may identify.

To implement this prioritization, ACRS Subcommittee Chairs will engage with staff to assess the importance of topics posed for review based on the above criteria and make a recommendation to the Full Committee as to whether a topic warrants our in-depth review.

We have already identified several areas in which our in-depth reviews may no longer be needed, such as requests for power uprates less than 7 percent, requests for plants to operate in the expanded power to flow domain, and routine license renewal applications.

We are currently exploring with the staff a more effective approach to our ongoing review of the NuScale design certification application (DCA). ACRS has historically conducted Phases 3 and 5 of Part 52 reviews on a chapter-by-chapter basis to identify and resolve technical issues. Under the proposed approach we would conduct our Phase 5 NuScale DCA review focused on key, risk-significant issues that are cross-cutting over multiple DCA chapters. This approach will emphasize technical integration and consistency in the design and will facilitate a more efficient and effective review.

To stay abreast of the agency's transformation initiatives, we will arrange for periodic updates. We will also continue to perform introspective evaluations to identify ways to improve our own effectiveness and efficiency and to contribute to agency transformation activities.

As we endeavor to further enhance our effectiveness and efficiency, we will also heed the words of one former commissioner: "The greatest value of the ACRS is its mere existence. Both the staff and the industry work harder to produce quality work when they know that they have to present their proposals to the Committee." In our reviews and independent advice, we will continue to provide the depth and breadth necessary to maintain these expectations.

Our broad collective expertise provides unique perspectives that can act as a proactive catalyst for the staff. Past examples include:

- Formative discussions that led to important approaches in risk-informed decision making (Regulatory Guide 1.174).
- An integrated approach to our review of NRC research programs to help prioritize future activities.
- Strong advocacy of a principle-based regulatory approach to digital I&C architectures based on independence, diversity, redundancy, deterministic processing, and access control. The staff has used this in digital I&C reviews.

We anticipate that similar interactions will lead to practical, risk-informed approaches in the Licensing Modernization Program and additional flexibility under 10 CFR 50.59.

ACRS Staffing and Budget

The agency is moving forward with efforts to optimally position itself to accomplish its mission in a changing regulatory framework and culture, and industry environment. Over the past several years, we have collaborated with NRC program offices to achieve greater efficiencies while maintaining our independence. Our operating costs have seen significant reductions and continue to trend downwards. We will continue to seek approaches to provide technical advice and support to the Commission in the most beneficial and efficient manner.

SUMMARY AND CONCLUSIONS

The focus of our actions to support the agency's transformation efforts, as described in this letter report, will be to prioritize our review and advisory activities to maximize those which will have the most impact and value to the Commission, while at the same time, maintaining our independence and technical competence that has been an asset to both the agency and the industry.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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October 17, 2019

SUBJECT: ACRS ACTIVITIES TO SUPPORT NRC TRANSFORMATION

Accession No: ML19290F956 **Publicly Available Y** **Sensitive N**
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

October 7, 2019

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

SUBJECT: REVIEW OF DRAFT SECY PAPER, "POPULATION - RELATED SITING CONSIDERATIONS FOR ADVANCED REACTORS"

Dear Chairman Svinicki:

During the 666th meeting of the Advisory Committee on Reactor Safeguards, September 4-6, 2019, we reviewed the draft SECY Paper entitled, "Population-Related Siting Considerations for Advanced Reactors." Our Future Plant Designs Subcommittee also reviewed this matter during a meeting on August 23, 2019. During these meetings we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATIONS

1. The paper provides four options for revising siting considerations for advanced reactors. We agree that Option 3 is the most reasonable of these approaches. However, it is short on details of implementation that will determine its ultimate value.

2. While Option 3 is short on details, they should be provided for review in the revised Regulatory Guide (RG) 4.7 with appropriate illustrative examples.

BACKGROUND

The NRC staff developed a report in 2016, "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Mission Readiness," as the staff contemplated how to review and regulate a new generation of non-light water reactors (non-LWRs), including their fuel cycles and waste forms. The report lays out six strategies to accomplish its goals and provides a set of Implementation Action Plans that identify specific, actionable tasks, which can fulfill the strategies. Over the intervening years, we have provided letter reports on the vision and strategy document as well as several products of the Implementation Action Plans—non-LWR design criteria, functional containment performance criteria, and the licensing modernization project (LMP or now called the "Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors").

The draft SECY paper, "Population-Related Siting Considerations for Advanced Reactors," was prepared under Strategy 5: identify and resolve technology-inclusive policy issues that impact

regulatory reviews, siting, permitting, and/or licensing of non-LWR nuclear power plants. It is intimately related to most of the previously reviewed documents. All these issues, especially source term characterization, functional containment performance, siting, and emergency planning are interdependent.

The subject of reactor siting has a long history of technical considerations and regulatory policy that began in the 1950s, when pressure was growing to site plants closer to power producers' customers. It became important to rely more on containment than isolation and simultaneously to develop siting criteria that would protect those populations. Today we have more than 60 years of institutionalized expectations, based on large LWR experience, analyses, and bounding approximations.

DISCUSSION

THE DRAFT SECY

The staff has identified the issue of siting decisions related to nearby populations as a matter that warrants early engagement with the Commission. During development of the draft SECY the staff considered possible changes to population-related siting considerations appropriate for advanced reactors, which can have substantially different design and accident characteristics than existing LWRs.

The paper is written at a high level, leaving many details unaddressed. The viability of the implementation of any of their proposed options will depend heavily on the details. We expect these details to be included in an associated revision to RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations." The paper itself becomes a bit hard to track, with a number of dense footnotes that contain essential information that provides technical justification for the approach. We suspect this came about in an attempt to retain a high-level, policy focus. Unfortunately, it also obscures much of the message.

The staff found that no changes to regulations would be required to satisfy their goal. They propose four options. One that maintains the status quo, and three more that would require changes to the guidance provided in RG 4.7: the second simply applies a source term factor based on power level, the third is based on dose calculations, and the fourth proposes developing societal risk measures.

One specific caveat not raised in the draft SECY, but implied in all the licensing activities for new non-LWR designs flowing out of the vision and strategy process, is the need for examining new designs with a clean sheet of paper. Improvements in our ability to calculate source terms and consequences in conjunction with the inherent safety aspects of advanced designs can reduce the probability and consequences of many of the events that have historically dominated the risk at LWRs. Nevertheless, one must be sure to think carefully about the failures and combinations of failures that could occur; i.e., what could go wrong. There are many tools that can help in such a search: a simple reframing—asking 'how could I make this system fail'; employing a search scheme similar to the Hazard and Operability Study (HAZOP) approach used in the chemical processing industry; and applying a modified failure modes and effects analysis at the system level rather than at the component level.

There is a tendency to believe in the perfection of new designs, especially when they are developed to eliminate the dominant failure scenarios in existing designs. However, one must remain vigilant and remember that nature provides surprises. There will be new accident

scenarios and new combinations of events to be considered that challenge our expectations and our assumptions about these advanced reactor systems. Creative thinking will be required to identify such unique situations, to thoroughly identify the scenarios that will be the basis of the safety analysis and the source of releases, and to evaluate the suitability of sites.

THE SITING REGULATIONS

The siting regulations are brief and clear. They are provided in 10 CFR Parts 50, 52, and 100. The regulations define an exclusion area (EA), a low population zone (LPZ), and a population center distance. The boundaries of the EA and LPZ are set by dose limits of 25 Rem Total Effective Dose Equivalent (TEDE) over the most limiting 2 hours for the EA and over the entire passage of the radioactive cloud for the LPZ. The plant must be sited a distance at least 1.33 times the radius of the LPZ from the boundary of any densely populated center of more than 25,000 people.

EXISTING POPULATION GUIDANCE

Siting guidance is provided in RG 4.7. It adds a population density criterion to the requirements of the regulations. "A reactor should be located so that, at the time of initial plant approval and within about 5 years thereafter, the population density, including weighted transient population, averaged over any radial distance out to 20 miles (cumulative population at a distance divided by the circular area at that distance), does not exceed 500 persons per square mile (ppsm). A reactor should not be located at a site where the population density is well in excess of this value."

STAFF OPTIONS IN THE DRAFT SECY

While deciding to maintain siting and population considerations as an element of defense in depth for future reactors, the NRC has recognized for many years that the specific source term and siting practices used for large LWRs may not be appropriate for the licensing and regulation of advanced reactor designs. For large LWRs dose calculations are based on an assumed set of severe accidents with substantial degradation of the core and the subsequent large release of radioactive material from the plant. Factors such as smaller source terms (fission product release to the environment), passive safety systems, and advances in barrier technology may allow for siting such reactors closer to densely populated centers than has historically been accepted for large LWRs.

The staff has identified two primary issues for the possible deployment of advanced reactors and the NRC's current siting requirements and guidance. The first issue involves the current limitations in RG 4.7 on population density to not exceed 500 ppsm out to a distance of 20 miles from a reactor site. This provision might unnecessarily preclude many sites associated with retiring fossil plants or industrial sites with relatively large population centers closer than 20 miles. The second issue involves the potential use of small modular reactors (SMRs) and microreactors for remote communities or smaller grids with relatively small but concentrated populations that would be near a reactor site.

The draft SECY paper describes four options the NRC staff has developed following a series of public meetings at which attendees discussed modifying the NRC rules and guidance to accommodate expected advances in reactor designs.

Option 1 is to maintain the status quo with no changes to the current population-related siting regulations or the existing guidance in RG 4.7.

This option gives no consideration to the expected improvements in new designs that eliminate vulnerability to some initiating events and limit source terms thereby reducing the likelihood and severity of most accidents. Option 1 does not reduce regulatory uncertainty and goes against the Commission goal to minimize complexity and add stability.

Option 2 would revise the population-related guidance in RG 4.7 to include provisions for advanced reactor designs and more specifically for SMRs using a scaling of the source term with power level.

We find that Option 2, while providing some benefit, is arbitrary and does not account for other important specific design attributes of new reactors.

Option 3 would revise the population-related guidance in RG 4.7 to include additional provisions for advanced reactor designs. The criteria are directly related to estimates of radiological consequences from design-specific events.

We find that Option 3 is attractive on several counts: it appears reasonable; it directly accounts for specific characteristics of each new design; and the criteria are directly related to calculated estimates of radiological consequences. It retains established principles, while allowing consideration of new reactor characteristics. Application of Option 3 will require substantial effort in identifying licensing basis events for evaluation and in developing mechanistic source terms to support dose calculations. Perhaps bounding simplifications for each source term may be possible.

Although calculable, the essence of the change is summarized in the paper as "The proposed criterion is that the population density (500 ppsm) would be assessed out to a distance equal to twice the distance at which a hypothetical individual could receive a calculated dose of 1 rem over a period of 1 month from the release of radionuclides resulting from the subject event categories." At first glance, this criterion seems to be deterministic rather than risk informed. In discussions with the staff that helped explain the footnotes in the draft SECY, it became clear that this criterion, based on dose, time and distance, does have risk-informed elements. First, the dose (1 rem) and time (1 month) are consistent with the required approach outlined in the LMP and DG-1353, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," for characterization of the dose from proposed advanced reactor designs. Further, the prescribed distance of twice the 1-rem dose boundary is approximately consistent with the distance required for relocation of local populations to satisfy the US Environmental Protection Agency Protective Action Guides. This approach seems reasonable, however, the use of footnotes to explain it to the reader made it quite inscrutable to the committee. This should be rectified in revisions to the final SECY.

The draft SECY permits an alternative to using the approach of the LMP and DG-1353. Applicants could take the traditional LWR approach using a stylized set of design basis accidents (DBAs) (still developed carefully for the new design), conservative single failure assumptions, and conservative source term in a manner similar to guidance provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear

Power Plants: LWR Edition," and RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," but adapted to their new designs.

Option 4 calls for the NRC staff to develop societal risk measures for assessing specific advanced reactor designs at specific sites. This option could be pursued by including the assessment of societal risks in RG 4.7 as an alternative to the current criteria on population density.

We find the idea of Option 4 laudable. It has been considered in the past and societal risk can yield different results than individual risk as witnessed by the events at Fukushima. Unfortunately, this could be difficult to develop into a widely accepted form and could devolve into interminable argument. Perhaps land contamination can be a suitable surrogate for societal risk, and it is calculated in full scope risk assessments. It could be added to the considerations in Option 3, provided that criteria could be developed that meet wide acceptance—perhaps a daunting task.

The staff recommends in the conclusions of the draft SECY Paper that the Commission approve Option 3, which consists of revising guidance to provide performance-based criteria to assess population-related issues in siting advanced reactors.

SUMMARY

The draft SECY paper provides four options for revising siting considerations for advanced reactors. We agree that Option 3 is the most reasonable of these approaches. Implementing it will require substantial work in identifying licensing basis events for evaluation and in developing mechanistic source terms for dose calculations. The preferred approach is technology-inclusive and risk-informed. There appears to be nothing that is focused on non-LWR aspects of design, and we see nothing that should preclude its use for LWR-based designs. It provides design-specific results and should be preferred technically to the present one-size fits all approach.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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October 7, 2019

SUBJECT: REVIEW OF DRAFT SECY PAPER, "POPULATION - RELATED SITING CONSIDERATIONS FOR ADVANCED REACTORS."

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

October 7, 2019

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE SUBSEQUENT LICENSE RENEWAL APPLICATION OF THE TURKEY POINT NUCLEAR GENERATING UNITS 3 AND 4

Dear Chairman Svinicki:

During the 666th meeting of the Advisory Committee on Reactor Safeguards (ACRS), September 4-6, 2019, we completed our review of the subsequent license renewal application (SLRA) for Turkey Point Nuclear Generating Units 3 and 4 (Turkey Point), and the associated final safety evaluation report. Our review considered actions by Florida Power and Light Company (FPL) to extend the license of each unit by 20 years beyond 60, thus becoming the first plant in the U.S. to apply for subsequent license renewal.

To conduct a focused review of past, current, and future actions to address subsequent license renewal at Turkey Point, our Plant License Renewal Subcommittee reviewed this matter during a meeting on June 21, 2019. During these reviews, we had the benefit of discussions with representatives of the staff and FPL. We also had the benefit of the referenced documents. This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

1. The programs established and the commitments made by FPL to manage age-related degradation provide reasonable assurance that Turkey Point can be operated in accordance with its licensing basis for the subsequent period of extended operation without undue risk to the health and safety of the public.

2. The FPL application for subsequent license renewal of the operating license for Turkey Point should be approved.

BACKGROUND

Turkey Point Nuclear Generating Units 3 and 4 are located in Miami-Dade County, east of Florida City, FL. Each unit consists of a Westinghouse pressurized-water reactor with licensed thermal power of 2,644 MWt, with a corresponding gross electrical output of approximately

913 MWe and 923 MWe, respectively. The NRC issued the initial operating licenses on July 19, 1972, for Unit 3 and April 10,1973, for Unit 4. The NRC issued the first renewed operating licenses on June 6, 2002.

In this application, FPL requests renewal of the operating licenses for an additional 20 years beyond the expiration of their current renewed licenses. The licenses would be extended to July 19, 2052, for Unit 3 and to April 10, 2053, for Unit 4.

DISCUSSION

The staff reviewed the FPL License Amendment Request for Subsequent License Renewal (SLR) in accordance with the Generic Aging Lessons Learned (GALL)-SLR and the Standard Review Plan (SRP)-SLR guidance documents. Conformance with this guidance provides bases for a conclusion that an applicant for life extension of 20 additional years beyond 60 years will assure adequate protection to the public through the Subsequent Period of Extended Operation (SPEO).

The most significant generic issues challenging operation beyond 60 years are: reactor pressure vessel embrittlement; irradiation-assisted stress corrosion cracking of reactor internals; concrete structures and containment degradation; and electrical cable environmental qualification, condition monitoring, and assessment. Each of these items has been addressed by FPL and evaluated by the staff through the review process. We agree with the staff's safety evaluation report regarding these issues.

Since our review of the Turkey Point SLRA, new information has been identified regarding the Regulatory Guide 1.99 irradiation embrittlement correlations that suggests they may be inaccurate at high fluence levels such as those expected to be experienced at the Turkey Point Reactor Pressure Vessels (RPVs) as they approach the end of their SPEO. While not an immediate concern for the Turkey Point units, staff and FPL should follow these developments and adjust their RPV irradiation embrittlement Aging Management Program (AMP) accordingly.

In preparation for life extension, FPL completed improvements, upgrades, replacements, and modifications to numerous systems and components. Significant plant modifications since initial license renewal include replacing reactor vessel heads, main and auxiliary transformers, the cask crane structure and crane, high pressure turbine rotors, main condenser tube bundles and water boxes, turbine plant cooling water heat exchangers, and condensate pumps. FPL also performed rehabilitation of the cooling canal and added two emergency diesel generators. Plant modifications currently in progress are replacement of low pressure turbine rotors, containment spray piping replacement, and modifications to the plant structure.

In its final safety evaluation report, the staff documented its review of the SLRA and other information submitted by FPL and obtained through staff audits and inspections at the plant site. The staff reviewed the completeness of the identification of structures, systems, and components that are within the scope of license renewal.

FPL will implement 50 AMPs for license renewal, comprised of 36 existing programs and 14 new programs. Of the 14 new programs, 12 are consistent with the GALL-SLR Report, one is consistent with enhancement, and one is plant specific (high-voltage insulators). Of the 36 existing programs, 24 are consistent with enhancements, one is consistent with allowed exceptions, 10 are consistent with enhancements and allowed exceptions, and one is plant

specific (Pressurizer Surge Line Fatigue Program). The SLRA includes eleven programs with allowed exceptions to the GALL-SLR Report. The programs with exceptions and enhancements are acceptable.

FPL has demonstrated the effectiveness of their programs to maintain material condition, to sustain system and equipment performance, and to identify improvements to assure facility safety and reliability. Knowledge transfer will be provided through formal mentoring by SLR program managers and their teams. FPL is implementing lessons learned from both their own license renewal experience as well as those from the industry fleet. Commitments in the SLRA and in FPL responses to the staff audits and inspections provide assurance that these programs will continue throughout the SPEO.

The staff conducted license renewal audits, and the audits verified the appropriateness of the FPL scoping and screening methodology for AMPs, the appropriateness of the aging management review, and the acceptability of the Time Limited Aging Analysis. The staff audit reports confirm the validity of the Turkey Point Aging Management Program. The Post-Approval Site Inspection for License Renewal verified that the license renewal requirements are implemented appropriately. The audits and inspection were comprehensive, and the corresponding reports were thorough.

Based on these audits, inspections, and the staff reviews, the staff concluded that FPL has demonstrated that the effects of aging at Turkey Point will be adequately managed. Safety functions will be maintained consistent with Turkey Point's licensing basis for the SPEO, as required by 10 CFR 54.21(a)(3). The staff's review of the SLRA identified no confirmatory items and one open item relating to the Buried and Underground Piping and Tanks AMP. This open item has subsequently been resolved with an accelerated cathodic protection program and enhanced condition inspections over the 10-year period prior to SPEO. We agree with the staff's conclusion that there are no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) that preclude renewal of the operating license for Turkey Point.

SUMMARY

The programs established and the commitments made by FPL to manage age-related degradation provide reasonable assurance that Turkey Point can be operated in accordance with its licensing basis for the SPEO without undue risk to the health and safety of the public. The FPL application for a SLR of the operating license for Turkey Point should be approved.

Members Riccardella and Sunseri did not participate in portions of the meeting related to fatigue of Class 1 components, environmentally assisted fatigue, and leak before break analysis in the application.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the Subsequent License Renewal of Turkey Point Generating Units 3 and 4," May 2019 (ML19078A012).

3. U.S. Nuclear Regulatory Commission, "Turkey Point Nuclear Generating Unit Nos. 3 and 4 - Plan for the Operating Experience Audit Regarding the Subsequent License Renewal Application Review (EPID No. L-2018-RNW-0002)," April 26, 2018 (ML18086A705).

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6. U.S. Nuclear Regulatory Commission, "Turkey Point Nuclear Generating Unit Nos. 3 and 4 - Report for the In-Office Regulatory Audit Regarding the Subsequent License Renewal Application Review (EPID No. L-2018-RNW-0002)," October 15, 2018 (ML18230B482).

7. U.S. Nuclear Regulatory Commission, "Turkey Point Nuclear Generating Unit Nos. 3 and 4 - Plan for the On-Site Regulatory Audit Regarding the Subsequent License Renewal Application Review (EPID No. L-2018-RNW-0002)," August 21, 2018 (ML18232A576).

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9. U.S. Nuclear Regulatory Commission, "Turkey Point Nuclear Generating Station – Nuclear Regulatory Commission Integrated Inspection Report 05000250/2019001 and 05000251/2019001," May 14, 2019 (ML19134A371).

10. Florida Power & Light Company, "Turkey Point Units 3 and 4 - Submittal of Updated Final Safety Analysis Report - Unit 4 Cycle 29 Update and License Renewal 10 CFR 54.37(b) Report," April 26, 2018 (ML18117A085).

11. U.S. Nuclear Regulatory Commission, NRC NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), December 2010 (ML103490036).

12. U.S. Nuclear Regulatory Commission, NRC NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," December 2010 (ML103490041).

13. U.S. Nuclear Regulatory Commission, NRC NUREG-2191, Volume 1, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report Final Report," July 2017 (ML17187A031).

14. U.S. Nuclear Regulatory Commission, NRC NUREG-2191, Volume 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," July 2017 (ML17187A204).

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17. U.S. Nuclear Regulatory Commission, NRC NUREG-2221, "Technical Bases for Changes in the Subsequent License Renewal Guidance Documents NUREG–2191 and NUREG–2192," December 2017 (ML17362A126).

18. U.S. Nuclear Regulatory Commission, NRC NUREG-2222, "Disposition of Public Comments on the Draft Subsequent License Renewal Guidance Documents NUREG–2191 and NUREG–2192," December 2017 (ML17362A143).

19. U.S. Nuclear Regulatory Commission, "Turkey Point Nuclear Generating Units 3 and 4 - Plan for the Irradiated Concrete Technical Issue Regulatory Audit Regarding the Subsequent License Renewal Application Review (EPID No. L-2018-RNW-0002)," July 5, 2018 (ML18173A087).

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October 7, 2019

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE SUBSEQUENT LICENSE RENEWAL APPLICATION OF THE TURKEY POINT NUCLEAR GENERATING UNITS 3 AND 4

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UNITED STATES
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WASHINGTON, DC 20555 - 0001

September 26, 2019

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

SUBJECT: SAFETY EVALUATION OF WESTINGHOUSE TOPICAL REPORT WCAP-17794-NP, REVISION 0, "10X10 SVEA FUEL CRITICAL POWER EXPERIMENTS AND NEW CPR CORRELATION: D5 FOR SVEA-96 OPTIMA3"

Dear Ms. Doane:

During the 666th meeting of the Advisory Committee on Reactor Safeguards, September 4-6, we reviewed the staff's safety evaluation report of Westinghouse Electric Company Topical Report WCAP-17794-P, Revision 0, "10X10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 OPTIMA3." Our review was also informed by presentations at April 18, 2019, and August 21, 2019, Thermal-Hydraulic Subcommittee briefings. During these meetings, we had the benefit of discussions with Westinghouse and the staff. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATION

1. The D5 critical power (CP) correlation, when used in compliance with the four limitations imposed by the staff, is acceptable for application to SVEA-96 Optima3 fuel.

2. The safety evaluation should be issued.

DISCUSSION

The D5 topical report documents a correlation to estimate the CP for SVEA-96 Optima3 fuel. Optima3 fuel introduces a few evolutionary changes to SVEA-96 Optima2 fuel. Spacer design is the main improvement that increases the margin to CP.

A series of tests were conducted in the FRIGG facility to develop a database for CP conditions for Optima3 fuel. Both steady state and transient data were collected and analyzed to generate the D5 correlation, which uses a different formulation than the approved D4.1 correlation. This new formulation results in a better fit to all the data by accounting more accurately for part-length rods, pin powers, and axial power shapes.

The staff has reviewed the topical report using the methodology described in NUREG/KM-0013, "Credibility Assessment Framework for Critical Boiling Transition Models." It provides a well-

structured and logical approach to the review of data-driven models. This approach provides consistency and completeness to this and future reviews. We were pleased to see that the staff considered the suggestion in our June 15, 2018, letter and published this methodology in a publicly available document. Future submittals will benefit from the predictability that this methodology provides by defining all the information expected in the submittal.

The staff has imposed four limitations on the use of this correlation. These limit the range of its applicability and ensure appropriate conservatisms in the unlikely event when bundles with high pin-power peaking become limiting.

In 2014, Optima2 fuel loaded in a foreign Boiling Water Reactor (BWR6) reactor showed signs of degradation in the form of V-shaped markings. The issue has been reviewed thoroughly by a multinational multi-disciplinary team, and the staff conclusion is that it does not affect the use of the D5 correlation for Optima3 fuel. Nevertheless, the staff imposed a limitation regarding the use of the D5 correlation in BWRs with lower plenum cross-beams as a defense-in-depth measure. We reviewed this issue in detail during an April 18, 2019, Subcommittee meeting and concur with the staff's evaluation.

SUMMARY

The D5 correlation, when used in compliance with the four limitations imposed by the staff, is acceptable for application to SVEA-96 Optima3 fuel. The safety evaluation should be issued.

Sincerely

/RA/

Peter Riccardella
Chairman

REFERENCES

1. Bergmann, U., Hemlin, M., Bergman, K., and J-M. Le Core, Westinghouse Electric Company, "10xl0 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 Optima3", WCAP-17794-P/NP, Revision 0, November 22, 2013 (ML13333A275 (Non-Proprietary Version/Publicly Available) and ML13333A276 (Proprietary Version/Non-Publicly Available)).

2. U.S. Nuclear Regulatory Commission, Safety Evaluation, "Draft Safety Evaluation for WCAP-17794-P, Revision 0, "10x10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 Optima3"," Revision 0, July 17, 2019 (ML19171A281 (Proprietary Version/Non-Publicly Available)).

3. Advisory Committee of Reactor Safeguards, 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 the NuScale Power, LLC Topical Report TR-0116-21012, Revision 1, June 15, 2018 "NuScale Power Critical Heat Flux Correlations." " (ML18166A303 (Publicly Available))

4. U.S. Nuclear Regulatory Commission, NUREG/KM-0013, "Credibility Assessment Framework for Critical Boiling Transition Models-Generic Safety Case to Determine the Credibility of Critical Heat Flux and Critical Power Models – Draft Report for Comment," March 31, 2019 (ML19073A249 (Publicly Available))

September 26, 2019

SUBJECT: SAFETY EVALUATION OF WESTINGHOUSE TOPICAL REPORT WCAP-17794-NP, REVISION 0, "10X10 SVEA FUEL CRITICAL POWER EXPERIMENTS AND NEW CPR CORRELATION: D5 FOR SVEA-96 OPTIMA3"

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UNITED STATES
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WASHINGTON, DC 20555 - 0001

September 25, 2019

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

SUBJECT: PROPOSED FOCUS AREA REVIEW APPROACH OF THE ADVANCED SAFETY EVALUATION REPORT WITH NO OPEN ITEMS FOR THE DESIGN CERTIFICATION APPLICATION OF THE NUSCALE SMALL MODULAR REACTOR

Dear Ms. Doane:

During the 666th meeting of the Advisory Committee on Reactor Safeguards, September 4-6, 2019, the Committee adopted a recommendation to conduct a focus area approach to review the advanced safety evaluation report (SER) with no open items associated with the NuScale design certification application (DCA) review. The Committee has completed its review of the SER with open items (Phase 3) and is anticipating receipt of the staff's advanced SER with no open items (Phase 4) on December 12, 2019, with the objective of completing our review (Phase 5) by June 23, 2020.

To meet these objectives, we propose to have our chapter leads conduct a completeness review of the advanced SER chapters. However, unless the review of an individual chapter warrants a subcommittee meeting, we would focus our attention on potentially safety-significant issues that are cross-cutting over multiple DCA chapters, as documented in our letter reports during Phase 3. The currently identified areas of focus include:

- Emergency Core Cooling System and Valve Performance;
- Helical Tube Steam Generator Design;
- Boron Dilution and Return to Criticality;
- Source Term; and
- Probabilistic Risk Assessment.

We recognize that this is a departure from past reviews of design certification applications; however, we see considerable merit in this approach in assuring a more complete, in-depth

review of matters that are inherently cross-cutting regarding integrated system safety performance. We request your support and the cooperation of the staff in planning and implementing this approach for the new calendar year.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

REFERENCE

1. U.S. Nuclear Regulatory Commission, "Review Schedule for the NuScale Power, LLC, Standard Design Certification of a Small Modular Reactor," May 22, 2017 (ML17103A380).

September 25, 2019

SUBJECT: PROPOSED FOCUS AREA REVIEW APPROACH OF THE ADVANCED SAFETY EVALUATION REPORT WITH NO OPEN ITEMS FOR THE DESIGN CERTIFICATION APPLICATION OF THE NUSCALE SMALL MODULAR REACTOR

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September 24, 2019

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

SUBJECT: SAFETY EVALUATION OF THE NUSCALE TOPICAL REPORT TR-0716-50351, REVISION 0, "NUSCALE APPLICABILITY OF AREVA METHOD FOR THE EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE TO EXTERNALLY APPLIED FORCES"

Dear Ms. Doane:

During the 666th meeting of the Advisory Committee on Reactor Safeguards, September 4-6, 2019, we reviewed the staff's safety evaluation report of NuScale topical report, TR-0716-50351, Revision 0, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces." Our NuScale Subcommittee also reviewed this topical report on August 20, 2019. During these meetings, we had the benefit of discussions with NuScale, Framatome, and the staff. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATION

1. The fuel assembly structural response methodology described in TR-0716-50351 is acceptable for use in performing NuScale fuel system structural response analyses. The associated safety evaluation report should be issued.

2. The modifications to the approved ANP-10337P-A methodology will ensure that the seismic analysis of the NuScale fuel will be in conformance with General Design Criterion 2; 10 CFR Part 50, Appendix S; and related staff guidance.

BACKGROUND

NuScale submitted a design certification application for its small modular reactor on December 31, 2016. Appendix 3A, Revision 2 of the application provides the seismic analysis of the NuScale Power Module (NPM). The NPM includes the reactor vessel, containment vessel, fuel and the associated structures, systems, and components.

NuScale submitted TR-0716-50351 on September 30, 2016, to be referenced as part of its design certification application. This topical report examines the applicability of the AREVA fuel assembly structural response analysis methodology. The NRC approved the Framatome (formerly AREVA)

topical report ANP-10337P-A, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations" for referencing in license applications for operating reactors on May 21, 2018. The methodology presented in ANP-10337P-A covers structural acceptance criteria, model architecture, model parameter and allowable limits definition, seismic and loss-of-coolant accident analysis, and non-grid component strength evaluation methodology.

The NuScale topical report evaluated the applicability of each section of the ANP-10337P-A report to the NuScale fuel assembly and plant design. Additionally, the report identified NuScale fuel design differences and potential analysis impacts.

DISCUSSION

The NuScale fuel assembly, NuFuel-HTP2™, is similar to the standard 17x17 Framatome HTP™ fuel design, with M5 fuel pin cladding, Zircaloy-4 guide tubes, HTP™ grids, and HMP™ bottom grids. However, NuFuel-HTP2™ is about one-half the length, and it contains 5 versus 7 spacer grids. To accommodate these differences, the NuScale topical report clarifies the ANP-10337P-A methodology in the following areas:

- methodology for evaluating fuel in the irradiated condition, including its effect on both spacer grids and overall fuel assembly structural response;
- spacer grid allowable impact load in both the irradiated and non-irradiated condition;
- protocol for benchmarking fuel assembly dynamic characteristics from tests;
- methodology for calculating non-grid component loads and stresses;
- acceptance criteria for guide tube stresses under loss-of-coolant accident and safe shutdown earthquake loads;
- description of the numerical model for vertical load analysis;
- methodology for combining loads from the horizontal and vertical analyses; and
- structural damping of the fuel assemblies due to NPM design differences.

SUMMARY

The fuel assembly structural response methodology described in TR-0716-50351 is acceptable for use in performing NuScale fuel system structural response analyses. The associated safety evaluation report should be issued. The modifications to the approved ANP-10337P-A methodology will ensure that the seismic analysis of the NuScale fuel will be in conformance with General Design Criterion 2; 10 CFR Part 50 Appendix S; and related staff guidance.

Sincerely,

/RA/

Peter Riccardella
Chairman

REFERENCES

1. U. S. Nuclear Regulatory Commission, "Advance Safety Evaluation on NuScale's TR-0716-50351, Revision 0, 'NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces'," August 21, 2019 (ML19233A071).

2. NuScale Power, LLC, TR-0716-50351, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," Revision 0, September 30, 2016 (ML16274A469).

3. Framatome, ANP-10337P-A, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," Revision 0, April 30, 2018 (ML18144A816).

September 24, 2019

SUBJECT: SAFETY EVALUATION OF THE NUSCALE TOPICAL REPORT TR-0716-50351, REVISION 0, "NUSCALE APPLICABILITY OF AREVA METHOD FOR THE EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE TO EXTERNALLY APPLIED FORCES"

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September 20, 2019

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

SUBJECT: SAFETY EVALUATION OF THE NUSCALE TOPICAL REPORT TR-0516-49417-P, REVISION 0, "EVALUATION METHODOLOGY FOR STABILITY ANALYSIS OF THE NUSCALE POWER MODULE"

Dear Ms. Doane:

During the 665th and 666th meetings of the Advisory Committee on Reactor Safeguards, July 10-12, 2019 and September 4-6, 2019, we reviewed the staff's safety evaluation report of NuScale topical report, TR-0516-49417-P, Revision 0, "Evaluation Methodology for Stability Analysis of the NuScale Power Module." Our NuScale Subcommittee also reviewed this topical report on June 19, 2019. During these meetings, we had the benefit of discussions with NuScale and the staff. We also had the benefit of the referenced documents.

CONCLUSION AND RECOMMENDATIONS

1. When used in compliance with the 16 limitations imposed by the staff, the methods documented in this stability topical report are acceptable for performing stability analyses of the NuScale power module (NPM). The safety evaluation should be issued.

2. Prototypical steam generator tests and scoping staff analyses show that two-phase density-wave flow oscillations inside the tubes are possible with the current design, which could challenge thermal fatigue limits. NuScale and the staff are aware of the issue and are committed to resolving it prior to completion of the review.

DISCUSSION

The NuScale stability topical report presents a thorough review of the possible instability modes that may affect the NPM. NuScale concludes that the dominant mode is the riser natural-circulation instability. The staff has reviewed the impact of these possible modes and agrees with this conclusion.

To properly model the unique features of the NPM and its stability response, NuScale developed a dedicated computer code. The PIM code models the core, riser, and steam

generators (SGs) using numerical methods that, given the experience with boiling water reactor (BWR) instability modelling, are known to be accurate for instability calculations. The staff has reviewed the PIM code and found it acceptable.

NuScale has performed 19 stability tests in their NuScale Integrated System Test facility at various power levels. These tests confirm that NuScale's normal operating conditions are stable. The PIM code shows good agreement with these experimental data which confirms that the most stable operating condition is full power. At very low power, the stability margin degrades, but the NPM remains stable as long as the core riser remains free of voids.

NuScale has imposed an exclusion region in the operating domain to ensure stable operation. This region is defined by maintaining a margin to boiling conditions in the core riser section of the NPM. A protection system trip will be implemented by comparing the core exit thermocouple temperature to the saturation temperature derived from the pressurizer pressure. The staff has reviewed this solution and found it acceptable.

In response to a request for additional information (RAI), NuScale presented results from the SIET-TF2 prototypical SG tests, which exhibited unstable two-phase density-wave flow oscillations in the secondary side; i.e., inside the tubes where boiling occurs. The NuScale SG configuration is unique because in most SGs boiling occurs outside the tubes, which results in lower pressure drops and tends to minimize flow oscillations. The SGs have tube-inlet flow restrictors designed to minimize the possibility of unstable oscillations; however, they were found to be effective for some, but not all, test conditions. NuScale has committed to resolve this issue to minimize the possible impact of thermal fatigue on the SG. If the design allows flow oscillations, movement of the boiling boundary would create temperature oscillations in the tubes, potentially inducing thermal fatigue. The staff reviewed the SIET-TF2 SG tests and performed scoping calculations with the current tube design. These indicate that the SG tube flow is likely to oscillate, potentially with large amplitude; NuScale stated that their calculations show different trends. This issue is under review. If this oscillatory behavior is confirmed, then the ASME Code calculations for the SG must account for it.

Tube-flow oscillations will affect the SG heat transfer and induce pulses of cold/hot water in the NPM core, which will result in power oscillations. However, oscillations in the secondary side are of relatively short period (3 to 10 seconds), which should have minor impact on the primary side, where oscillations have periods of 100 to 500 seconds. In addition, by the nature of the flow oscillation, each of the approximately 1000 SG tubes is expected to oscillate with a random phase; therefore, the cumulative impact on the primary side is greatly reduced by averaging over all the SG tubes.

The staff was concerned that control-system instabilities could drive oscillations of all tubes in-phase. NuScale contends that this phenomenon will be precluded by tuning of the control system during initial testing. After reviewing all the available information, the staff's safety evaluation finds that NuScale analyses give reasonable assurance that core thermal margins will be maintained during the worst-case allowable secondary-side flow oscillations. We concur with the staff evaluation.

Even though core integrity is not compromised, the staff should ensure that NuScale's solution to the SG tube flow oscillation issue minimizes the potential of SG tube fatigue. We also note that resolution of this issue cuts across multiple disciplines. In this case, the staff in charge of the stability review communicated with the mechanical engineering staff and both are following up the eventual solution. However, this serves as an example for other topics where

compartmentalization of the review by chapter and disciplines may result in issues not being addressed due to lack of proper or timely communications. We plan to conduct our final phase of the NuScale design certification application review using a more structured, multi-discipline process that complements the chapter-by-chapter reviews already completed. We will provide our proposal for this revised process under separate cover.

A validated calculational tool to estimate the stability of the SG secondary-side flow stability is needed to increase confidence in reliable operation of the NuScale design. NuScale is in the process of developing a tube-flow stability map using NRELAP5, after further benchmarking against the TF-2 test results. The staff should review these new analyses, and we look forward to further discussion on this topic.

SUMMARY

The staff's safety evaluation concludes that, when exercised in compliance with the 16 limitations, the methods documented in the stability topical report are acceptable for performing stability analyses of the NPM. We concur with the staff's evaluation, and it should be issued.

Sincerely,

/RA/

Peter Riccardella
Chairman

REFERENCES

1. U. S. Nuclear Regulatory Commission, "NuScale Power, LLC, Safety Evaluation for NuScale Topical Report TR-0516-49417-P, Revision 0, "Evaluation Methodology for Stability Analysis of NuScale Power Module," September 11, 2019 (ML19254C858).

2. NuScale Power, LLC Topical Report TR-0516-49417-P, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," Revision 0, July 31, 2106 (ML16250A851).

3. NuScale Power, LLC Response to NRC Request for Additional Information No. 9181 (e.RAI No. 9171) on NuScale Topical Report, "Evaluation Methodology for Stability Analysis of NuScale Power Module," TR-0516-49417, Revision 0, February 16, 2018 (ML18047A737).

September 20, 2019

SUBJECT: SAFETY EVALUATION OF THE NUSCALE TOPICAL REPORT TR-0516-49417-P, REVISION 0, "EVALUATION METHODOLOGY FOR STABILITY ANALYSIS OF THE NUSCALE POWER MODULE"

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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WASHINGTON, DC 20555 - 0001

August 2, 2019

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

SUBJECT: INTERIM LETTER – CHAPTERS 3, 6, 15 AND 20 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 665th meeting of the Advisory Committee on Reactor Safeguards, July 10-12, 2019, we met with representatives of NuScale Power, LLC (NuScale) and the NRC staff to review Chapter 3, "Design of Structures, Components, Equipment and Systems;" Chapter 6, "Engineered Safety Features;" Chapter 15, "Transient and Accident Analyses;" and Chapter 20, "Mitigation of Beyond-Design-Basis Events;" 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 June 18-20, 2019, and July 9, 2019. During these meetings, we had the benefit of discussions with NuScale and the NRC staff. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATIONS

1. The barrier analysis used for turbine missile protection is a different approach than previously accepted. We await the staff's review before commenting.

2. The emergency core cooling system (ECCS) valve test program currently underway is required to provide confidence for valve functionality and performance.

3. NuScale's power module (NPM) can experience a return-to-power under accident analysis assumptions, but does not violate any specified acceptable fuel design limits. This potential operational condition should be precluded in the long term.

4. We have not identified any additional major issues at this time for Chapters 3, 6, 15 and 20.

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 its bases for acceptance of the DCA, the SER chapters also identify the information a combined license applicant must provide.

Our Phase 3 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 DCA review. This letter addresses the staff's SER and the DCA for NPM for Chapter 3, Revision 1; Chapter 6, Revision 2; Chapter 15, Revision 2; and Chapter 20, Revision 2; as well as supplementary material, including responses to staff requests for additional information.

DISCUSSION

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

DCA Chapter 3 – Design of Structures, Components, Equipment and Systems

This chapter documents the analytical methods, testing procedures, tests and analyses that the applicant used to ensure the structural and functional integrity of the piping systems, mechanical equipment, reactor vessel, reactor internals, and their supports under static and dynamic loadings, including those caused by normal operation and postulated events. It addresses how the design conforms to General Design Criteria. Individual sections discuss conformance to the applicable criteria: seismic classification of systems, structures and components (SSCs) important to safety; the analyses and tests performed to demonstrate acceptability of the SSCs under bounding seismic load spectra; and the capability of the NuScale design to withstand wind and tornado loadings and floods. In their review, the staff concludes that the design meets the applicable regulations in these areas.

Section 3.5 of the SER addresses protection of safety-related SSCs from missiles. Missiles generated by a turbine failure could potentially impact the Reactor Building and the Control Building. NuScale analyzes missiles impacting the Reactor Building concrete wall and the Control Building wall and grade slab to demonstrate that these are effective barriers to protect critical SSCs and related safety functions within those buildings. This is a different approach than has been previously licensed for turbine missile protection. At this time, the staff has not completed their review of additional information submitted by the applicant in response to staff requests on this topic, and it remains an open item. Our review and comment of this topic will await completion of that effort.

In our June 19, 2019, Subcommittee meeting with respect to the NuScale methodology for stability analysis of the NPM, we noted that there may be two-phase density-wave flow oscillations which would cause thermally induced fatigue of the steam generator tubes. NuScale and the staff have agreed to assure that these oscillations do not compromise the design limits and challenge tube integrity.

Other chapter sections address Protection against Dynamic Effects Associated with Postulated Pipe Rupture (including Leak Before Break); Design of Category I Structures; Environmental Qualification of Mechanical and Electrical Equipment; and ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports. The staff concludes that, pending resolution of confirmatory items, the NuScale design meets the applicable requirements in these areas.

DCA Chapter 6 – Engineered Safety Features

This chapter discusses the engineered safety systems that are part of the NPM: specifically, the containment systems, the ECCS, control room habitability, fission product removal and control systems, and in-service inspection and testing of systems and components.

The ECCS includes five valve systems, three of which are reactor-vent-valves mounted on the reactor pressure vessel upper head that are directly connected to the pressurizer steam space and discharge to containment, and two reactor-recirculation-valves mounted on the side of the reactor pressure vessel in the downcomer that open to the containment. All five valves are closed during normal plant operation and open following the receipt of an actuation signal resulting from applicable accident conditions. Those valves also open on a loss of direct current (DC) power, and therefore are not dependent on a power source for actuation.

The ECCS valves are sophisticated in their design in that they incorporate a DC powered solenoid trip valve for actuation in combination with an integrated hydraulic and spring mechanical valve to allow flow from the reactor pressure vessel to containment for blowdown and depressurization. Inadvertent ECCS valve actuation is one of the anticipated operational occurrences (AOOs) analyzed in Chapter 15. The staff is currently reviewing the ability of this system to perform its safety function. NuScale is in the process of conducting valve testing to demonstrate compliance with the regulatory requirements. Successful completion of this ECCS valve test program is required to provide confidence that the valves will function as designed.

DCA Chapter 15 – Transient and Accident Analyses

This chapter discusses the set of design basis events (DBEs) that the NPM, as designed and constructed, must withstand without loss of SSCs needed to maintain core cooling and containment integrity.

The staff presented 11 unresolved issues without a clearly defined, mutually understood path towards resolution. The final status of the review is based on resolution of these unresolved issues, as well as the remaining open items.

In addition, the loss-of-coolant accident (LOCA) and Non-LOCA topical report methodologies, which are the basis for the Chapter 15 analyses, are also under review. Completion of these reviews is treated as an open item. We note that the above two methodologies yield significantly different estimates for minimum critical heat flux ratio at steady-state conditions. The staff should ensure that these differences are resolved before approving these methodologies.

Design basis events (DBEs) are analyzed until the module protection system actuation places the reactor in a stable shutdown condition, typically achieved within a few seconds to a few minutes of event initiation. Long-term cooling is evaluated generically for all events. This generic evaluation includes combinations of alternating current (AC) and DC power availability (available or unavailable). This multiple-scenario evaluation identifies the worst-case conditions when no credit is taken for nonsafety-grade AC or DC power.

Long-term cooling of the NPM can be achieved by passive safety system features, including the decay heat removal system (DHRS) and the ECCS. Both systems are enabled by valves that fail safe on loss of power. Proper, reliable actuation of these valves is crucial to guarantee

passive long-term cooling. As noted previously, NuScale is conducting an important series of tests for the ECCS valves to demonstrate functionality and performance.

ECCS valve actuation relies on detecting high water level in the containment. NuScale plans to use a sensor never used under comparable conditions in the nuclear power industry. The staff should ensure that the level instrument is qualified to continuously measure the empty dry containment level, as well as accurately measure water level, when water enters the containment.

Chapter 15 assumes that some nonsafety-grade components may be used as backup for safety-grade components with similar function, e.g., main steam isolation valves and main feedwater valves. The staff has evaluated this type of event and concurs with the NuScale conclusion that this implementation is consistent with prior pressurized water reactor applications and is acceptable.

The staff has identified an open item related to the physical mechanisms that may result in non-uniform distribution of dissolved boron in the reactor vessel and containment after ECCS valve actuations. These mechanisms need to be understood to ensure that boron dilution in the core is limited and that subcriticality is maintained.

As part of the generic long-term cooling evaluation, NuScale has analyzed the possibility of return-to-power events. Several concurrent conditions must be present to lose shutdown margin: (1) the highest-worth control rod fails to insert on demand and remains withdrawn; (2) large moderator temperature reactivity feedback, which typically occurs close to the end of cycle; (3) low decay heat, otherwise the resulting small steam void levels would prevent criticality; and (4) significant overcooling following a reactor trip, resulting in temperatures lower than expected by either normal operator control or passive DHRS actuation. Both the applicant and the staff have performed an extensive evaluation of this type of event, including multiple sets of initial conditions and reactor parameters. These evaluations confirm that return to power is possible, and the expected power levels are small, generally on the order of 2 percent of nominal power. At this power level, core-coolability criteria are satisfied with sufficient margin. This event is the subject of an exemption request to General Design Criterion (GDC) 27, which is currently under review and consideration by the staff.

The standard review plan recommends adherence to conservative specified acceptable fuel design limits (SAFDLs) for AOOs, and it allows less stringent fuel design limits for lower frequency postulated accidents such as a LOCA. NuScale has chosen to apply the more conservative SAFDLs for both AOOs and postulated accidents to ensure compliance with the staff requirement that the SAFDL criteria (e.g., not exceeding critical heat flux under any circumstances) be applied to any event that may result in a return-to-power because of uncertainties in the event progression in the long term.

These analyses demonstrate that a return-to-power is highly unlikely; but, it is a situation that could leave the NPM in a critical, low power state in the long term. While these analyses predict the fuel remains within the SAFDL requirements for core coolability, this situation should not be allowed to persist or be considered as an acceptable final stable state for an NPM.

Consistent with the approach where nonsafety-grade components are used to backup failures in safety-grade components of similar function, measures should be taken in the NPM design and operation to preclude this situation in the long term (i.e., past 72 hours, when Chapter 15 rules do not apply) by giving credit to nonsafety-grade boron addition equipment.

The staff is continuing its review and is working with the applicant on a resolution path for all open items to ensure that all analyzed AOOs and postulated accidents satisfy the conservative SAFDL criteria.

DCA Chapter 20 – Mitigation of Beyond Design Basis Events (MBDBE)

This chapter outlines NuScale strategies to address the requirements in the pending MBDBE rule, 10 CFR 50.155. In addition, this chapter discusses NuScale's plans to address existing 10 CFR 50.54(hh)(2) requirements for loss of large area due to fire or explosion and associated procedure integration as well as emergency response planning. Design certification applicants can defer all MBDBE requirements to the combined license applicant. However, to reduce regulatory uncertainty for the combined license applicant, NuScale is voluntarily seeking NRC approval for its strategies to meet these requirements in the DCA.

In their review, the staff agreed with NuScale's proposed plans regarding 10 CFR 50.54(hh)(2) loss of large area, procedure integration, and emergency response. However, two open items precluded the staff from making a safety finding with respect to pending 10 CFR 50.155 requirements – the mechanisms responsible for flow fluctuations observed in DHRS operation and the potential for recriticality due to boron redistribution. The staff expects that on-going interactions will resolve these items.

The pending rule requires mitigation strategies for beyond design basis external events that are developed assuming a loss of all AC power concurrent with a loss of normal access to the ultimate heat sink or a loss normal access to the normal heat sink. Because of the enhanced safety margins or increased reliance on passive measures in new designs, it is reasonable to propose mitigating strategies that differ from those implemented in operating reactors. NuScale contends that their design is sufficiently robust to prevent damage to fuel in any NPM and the spent fuel pool and to maintain containment function for greater than 30 days. Consistent with prior applications, the staff has limited their review to the performance of permanently installed SSCs for 72 hours following a BDBE.

<u>SUMMARY</u>

The barrier analysis used for turbine missile protection is a different approach than previously accepted. We await the staff's review before commenting. The ECCS valve test program currently underway is required to provide needed confidence for valve functionality and performance. NuScale's power module can experience a return-to-power under accident analysis assumptions, but does not violate any SAFDLs. This potential operational condition should be precluded in the long term. We have not identified any additional major issues at this time for Chapters 3, 6, 15 and 20.

Members Riccardella and Sunseri did not participate in Chapter 3, Section 3.5 deliberations.

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 3, 'Design of Structures, Systems, Components and Equipment'," May 21, 2019 (ML19140A381).

2. NuScale Power, Design Certification Application, Chapter 3, "Design of Structures, Systems, Components and Equipment," Revision 1, March 15, 2018 (ML18086A153-171).

3. U. S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application - Safety Evaluation With Open Items for Chapter 6, 'Engineered Safety Features'," May 20, 2019 (ML19130A046).

4. NuScale Power, Design Certification Application, Chapter 6, "Engineered Safety Features," Revision 2, October 30, 2018 (ML18310A327).

5. U. S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application – Safety Evaluation With Open Items for Chapter 15, "Transient and Accident Analyses," July 9, 2019 (ML19168A103).

6. NuScale Power, Design Certification Application, Chapter 15, "Transient and Accident Analyses," Revision 2, October 30, 2018 (ML18310A337).

7. U. S. Nuclear Regulatory Commission, "NuScale Power, LLC Safety Evaluation for Topical Report TR-0516-49417-P, Revision 0, 'Evaluation Methodology for Stability Analysis of the NuScale Power Module'," May 15, 2019 (ML19135A412).

8. NuScale Power, Topical Report TR-0516-49417-P, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," Revision 0, July 31, 2016 (ML16250A851).

9. U. S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application – Safety Evaluation With Open Items for Chapter 20, "Mitigation of Beyond-Design-Basis Events," July 9, 2019 (ML19190A188).

10. NuScale Power, Design Certification Application, Chapter 20, "Mitigation of Beyond-Design-Basis Events," Revision 2, October 30, 2018 (ML18310A343).

August 2, 2019

SUBJECT: INTERIM LETTER – CHAPTERS 3, 6, 15 and 20 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE DESIGN CERTIFICATION APPLICATION REVIEW OF THE NUSCALE SMALL MODULAR REACTOR

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

June 27, 2019

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

SUBJECT: RESPONSE TO CHAPTER 2, "SITE CHARACTERISTICS AND SITE PARAMETERS," AND CHAPTER 17, "QUALITY ASSURANCE AND RELIABILITY ASSURANCE," OF THE U.S. NUCLEAR REGULATORY COMMISSION STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE NUSCALE POWER, LLC, SMALL MODULAR REACTOR

Dear Ms. Doane:

During the 664th meeting of the Advisory Committee on Reactor Safeguards, June 5-7, 2019, we reviewed the staff's April 17, 2019, letter regarding disposition of Conclusion 4 in our February 21, 2019, Interim Letter concerning Chapter 2, "Site Characteristics and Site Parameters," and Chapter 17, "Quality Assurance and Reliability Assurance," of the safety evaluation report with open items associated with the NuScale design certification application.

The Committee's interim letter included a recommendation that the applicant's Open Design Items (ODIs) for structures, systems and components covered by Chapter 17 requirements needed to be identified for eventual closure. The April 17, 2019, response to this recommendation from the Director of the Office of New Reactors described the role of the quality assurance program in assuring closure and concluded that further identification of ODIs was not necessary. Within the context of the NuScale review, ODIs are unverified design assumptions.

The Committee requested an informational meeting with the staff to better understand the generic process for ensuring subsequent closure of the set of unverified design assumptions that remain open at the time of design certification and are important to the finding of reasonable assurance of adequate protection. This set may or may not be within the scope of the quality assurance program described in Chapter 17 and represent only a small portion of what NuScale identifies as ODIs. This informational meeting was held with the Committee on June 6, 2019.

Based on the information provided by the staff during the meeting, 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.

We appreciate the responsiveness of staff to our questions concerning their review process.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

REFERENCES:

1. Advisory Committee on Reactor Safeguards, "Interim Letter: Chapter 2 and 17 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the NuScale Small Modular Reactor," February 21, 2019 (ML19052A046).

2. U.S. Nuclear Regulatory Commission, "NRO Response to ACRS Letter on Safety Evaluation of the NuScale Power LLC, Chapter 2 and 17," April 17, 2019 (ML19072A215).

June 27, 2019

SUBJECT: RESPONSE TO CHAPTER 2, "SITE CHARACTERISTICS AND SITE PARAMETERS," AND CHAPTER 17, "QUALITY ASSURANCE AND RELIABILITY ASSURANCE," OF THE U.S. NUCLEAR REGULATORY COMMISSION STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE NUSCALE POWER, LLC, SMALL MODULAR REACTOR

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

June 20, 2019

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

SUBJECT: REVIEW OF NUCLEAR ENERGY INSTITUTE (NEI) 96-07, APPENDIX D, "SUPPLEMENTAL GUIDANCE FOR APPLICATION OF 10 CFR 50.59 TO DIGITAL MODIFICATIONS," DATED NOVEMBER 2018, 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"

Dear Chairman:

During the 664th meeting of the Advisory Committee on Reactor Safeguards, June 5-7, 2019, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to review the subject documents. Our Digital Instrumentation and Control (DI&C) Subcommittee reviewed these documents on April 16, 2019. These documents were issued for public comment on May 30, 2019.

CONCLUSIONS AND RECOMMENDATIONS

1. 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.

2. 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.

3. 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.

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

BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59, "Changes, Tests and Experiments," paragraph (c)(1) authorizes a licensee to make changes in the facility or procedures described in its UFSAR or perform tests or experiments not described in its UFSAR without obtaining a license amendment pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," only if (i), a change to the facility's technical specifications is not required and (ii), the change, test, or experiment does not meet any of the eight criteria in 10 CFR 50.59(c)(2). The NRC issued the final rule that adopted the eight criteria on October 4, 1999, and it took effect on March 13, 2001.

Nuclear Energy Institute issued NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," in November 2000, to aid the industry in developing the bases for determining if a license amendment request (LAR) was required for facility changes.

The NRC developed Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments, November 2000," endorsing NEI 96-07, Revision 1, as a method that the staff considers acceptable for use in complying with the NRC regulations on the process by which licensees, under certain conditions, may make changes to their facilities and procedures as described in the UFSAR, and conduct tests or experiments not described in the UFSAR, without prior NRC approval. NRC did not provide any clarifications or exceptions to the methods and examples in the NEI 96-07, Revision 1, guidance.

DISCUSSION

The main body of 10 CFR 50.59 is applicable for all SSCs. However, based on our discussions with the staff and representatives of NEI, we agree that expanded guidance for applying 10 CFR 50.59 to DI&C systems has been needed. There have been varying opinions on the interpretation of 10 CFR 50.59 requirements, when applying them to DI&C systems, as represented in submittals and discussions among the parties. These discrepancies have resulted 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. The new guidance provided in NEI 96-07, Appendix D; Regulatory Guide 1.187, Revision 2; and RIS 2002-22, Supplement 1, "Clarification on Endorsement of NEI Guidance in Designing Digital Upgrades in I&C Systems," should ensure that plant modifications performed by the licensee for DI&C systems or components without an LAR meet NRC expectations.

Both the RIS and NEI 96-07, Appendix D, are intended to assist licensees in the performance of 10 CFR 50.59 reviews of activities involving digital modifications. NEI 96-07, Appendix D, points to the RIS for guidance on qualitative evaluations and was submitted for endorsement by the NRC on January 19, 2019. NEI 96-07, Appendix D, is applicable to digital modifications involving safety-related and non-safety related systems and components and, also covers "digital-to-digital" activities. The RIS supplement is not directed towards DI&C replacements of the reactor protection system, the engineered safety features actuation system, or modification/replacement of the internal logic portions of these systems.

Draft Revision 2 to Regulatory Guide 1.187, endorses 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.

The safety case for a nuclear power plant is made by the complete UFSAR, not the safety analyses alone. A different result anywhere in the UFSAR could alter the basis for licensing of the plant. If that possibility arises, an LAR is required so that the staff will evaluate the effect of the change and associated change in the basis for licensing.

Given the current rule, the NEI focus on only the safety analysis is too narrow. A malfunction of a modified SSC or performance characteristic could have a not so obvious effect on other systems, procedures, or operator response actions. The fundamental issue is "Does a change, modification, or component or system replacement result in a change in the licensing basis for the plant?" This cannot be determined without a more complete review of the licensing basis rather than just the safety analysis when potential modifications are being considered.

However, the staff should consider possible approaches to allow more flexibility in the application of 10 CFR 50.59. The staff's exception to NEI 96-07, Appendix D, requires consideration of more than the safety analysis within the 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 an 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.

Sincerely,

/RA/

Peter Riccardella
Chairman

REFERENCES

1. Nuclear Energy Institute, NEI 96-07, "Guidelines for 10 CFR 50.59," Revision 1, November 17, 2000 (ML003771157)

2. Nuclear Energy Institute, NEI 96-07, Appendix B, "Guidelines for 10 CFR 72.48 Implementation," March 05, 2001 (ML010670023)

3. Nuclear Energy Institute, NEI 96-07, Appendix C, "Guideline for Implementation of Change Processes for New Nuclear Power Plants Licensed under 10 CFR Part 52," Revision 0, March 2014, (ML14091A739)

4. Nuclear Energy Institute, NEI 96-07, Appendix D, Supplemental Guidance for Application of 10 CFR 50.59 to Digital Modifications," Revision 0, November 2018 (ML18338A389)

5. U.S. Nuclear Regulatory Commission, Letter to NEI documenting staff comments on NEI 96-07, Appendix D, December 20, 2018 (ML18340A124)

6. U.S. Nuclear Regulatory Commission, Regulatory Issue Summary (RIS) 2002-22, Supplement 1, "Clarification on Endorsement of NEI Guidance in Designing Digital Upgrades in I&C Systems," Revision 1, May 31, 2018 (ML18143B633)

7. U.S. Nuclear Regulatory Commission, NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59 Changes, Tests, and Experiments," Revision 0, November 2000 (ML003759710)

8. U.S. Nuclear Regulatory Commission, NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59 Changes, Tests, and Experiments," Draft Revision 2, Draft Guide DG-1356, May 2019 (ML19045A435)

June 20, 2019

SUBJECT: REVIEW OF NUCLEAR ENERGY INSTITUTE (NEI) 96-07, APPENDIX D, "SUPPLEMENTAL GUIDANCE FOR APPLICATION OF 10 CFR 50.59 TO DIGITAL MODIFICATIONS," DATED NOVEMBER 2018, 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"

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UNITED STATES
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WASHINGTON, DC 20555 - 0001

June 19, 2019

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

SUBJECT: INTERIM LETTER – CHAPTER 3, SECTION 3.9.2, AND CHAPTERS 14, 19 AND 21 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 664th meeting of the Advisory Committee on Reactor Safeguards, June 5-7, 2019, we met with representatives of NuScale Power, LLC (NuScale) and the NRC staff to review 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," 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 May 14-16, 2019. During these meetings, we had the benefit of discussions with NuScale and the NRC staff. We also had the benefit of the referenced documents. Note that Chapter 19, Section 19.4, "Strategies and Guidance to Address Loss of Large Areas of the Plant Because of Explosions and Fire," of the DCA is evaluated as Section 20.2 by the staff and will be reviewed as part of Chapter 20 at a later date.

CONCLUSIONS AND RECOMMENDATIONS

1. 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).

2. We have not identified any major issues at this time for Chapter 3, Section 3.9.2, and Chapters 14, 19 and 21.

3. To help identify risk insights in this unique design, there are technical issues in the probabilistic risk assessment (PRA) that merit further consideration.

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 its 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. Our review addresses the staff's SER and DCA Chapter 3, Section 3.9.2, Revision 2; Chapter 14, Revision 2; Chapter 19, Revision 2; and Chapter 21, Revision 1, as well as supplementary material, including responses to staff requests for additional information.

DISCUSSION

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

DCA Chapter 3, Section 3.9.2 – Dynamic Testing and Analysis of Systems, Components and Equipment

This section of the Chapter 3 SER reviews 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.

In its review, the staff focused on dynamic system analysis of the reactor internals under service level D conditions and the applicant's reactor internals comprehensive vibration assessment program. In addition to Section 3.9.2, the staff review covered Appendix 3A of the DCA and four associated NuScale technical reports.

The staff identified several issues and open items in its review of NuScale's service level D dynamic system analyses. These concerned assumptions regarding system damping; fluid gap between the core barrel and reflector blocks; acoustic absorption coefficient for the reactor pool floor; generation of in-structure seismic response spectra; adequacy of NuScale power module (NPM) seismic analysis cases; uplift of reflector blocks; and seismic analysis details of major RVI components, including the steam generators. These items are either closed or in the process of being resolved.

Regarding NuScale's RVI vibration assessment program, the staff performed detailed evaluations of components considered most susceptible to flow-induced vibration (FIV) concerns, including the helical coil steam generator tubes and supports, the steam generator tube inlet flow restrictors, the control rod drive shafts, the in-core instrument guide tubes, and the NPM primary and secondary coolant piping.

Although the natural circulation NPMs have significantly lower primary coolant flow rates than conventional PWRs, some RVI components contain long rods and tubes that may be susceptible to FIV. Acoustic resonance may also be possible in dead piping legs adjacent to secondary coolant flow. The staff review identified some non-conservatisms in the NuScale FIV analyses that may outweigh the conservatisms. These are being addressed by testing. The

testing focuses on key FIV mechanisms with low margins of safety and high uncertainty, plus initial startup testing with online vibration monitoring performed in accordance with ASME standards. Prior to and following initial startup testing, components will be inspected for mechanical wear and signs of vibration-induced damage.

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 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.

DCA Chapter 14 – Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria

The initial test program consists of a series of preoperational and startup tests conducted by the startup organization. Preoperational testing is conducted for each 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.

The scope of the ITAACs addressing these tests needs to be sufficient to provide reasonable assurance that, if the ITAACs are successfully completed, the facility has been constructed and can be operated in accordance with the regulations, and the combined license. We concur with the staff that, pending completion of the confirmatory items and closure of the open items, NuScale has demonstrated compliance with the NRC regulations regarding its initial test program. In addition, we concur with the staff that the open items preclude finalization of conclusions related to ITAACs.

DCA Chapter 19 – Probabilistic Risk Assessment and Severe Accident Evaluation

Section 19.1

This section 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[1]. 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.

The PRA was integral to the design process, and risk insights influenced a number of design decisions. This integrated process contributed to achieving low values of core damage frequency (CDF) and large release frequency (LRF). These low values provide confidence that the NuScale design meets the Commission's Safety Goals with margin.

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[1] The PRA was performed for a single module. The likelihood to fail more than one module was approximated.

In fact, the NuScale PRA results indicate that the risk associated with the plant operation is apparently negligible. To further build confidence that these results accurately reflect the risk, we have identified the following technical issues that may merit further consideration by the staff.

Emergency Core Cooling System (ECCS) Valves Model and Data

Failures of NuScale's unique ECCS valves are among the most important risk contributors identified in the PRA. These valves are closed by hydraulic pressure and opened by spring actuation. The PRA failure model for these valves is not available in Chapter 19, but insights from some of the significant CDF cutsets suggest a possible non-conservative modeling of the valves passive opening at low differential pressure. We plan to visit NuScale for a further examination of the valve design and the associated PRA model to help build confidence that the plant risk is accurately represented.

Uncertainty Analysis

The applicant reported very low numerical values for the CDF and LRF. These values are based on available information but should have high uncertainty due to: (1) incomplete design and construction, undeveloped procedures, and a lack of operating experience as expected for any new design; and (2) lack of data on reliability of the design-unique and risk-significant components (e.g., the ECCS valves), or reliability of passive heat transfer to the reactor pool.

However, the results reported in Section 19.1 are not consistent with the expectation of high uncertainty. Even though the presented uncertainty results do not account for the model uncertainties, but only include parameter uncertainties, that alone would not explain these narrow uncertainty ranges. To ensure that the risk measures reported in the PRA include more realistic uncertainty results (and the associated mean values), the uncertainty analysis merits further investigation.

Sensitivity Analyses

Sensitivity analyses were performed to provide additional insights on the risk results and component importance measures, and to investigate the importance of modeling assumptions and uncertainties. As with uncertainty, these evaluations are especially important given the reported low numerical value for the risk measures. We found these sensitivity analyses incomplete in the following areas:

- No combinations of sensitivity results for different assumptions are included. The PRA results could be especially sensitive to a combination of specific assumptions and uncertainties in the same accident sequence.

- Some sensitivity cases are general and not event-specific; thus, their results may obscure relevant risk insights. For example, the sensitivity analysis on common cause failures (CCFs) selects a failure rate of 0.002 for all events and concludes that Safety Goals are still met. Nevertheless, the risk is not negligible in this case, and risk insights are different. A realistic sensitivity analysis could provide more meaningful insights; e.g., CCF for ECCS and decay heat removal system valves or degradation of passive cooling systems.

Based on the above discussion, more complete sensitivity analyses would increase confidence that the plant risk is accurately represented.

Human Errors of Commission

In the SER, the staff stated that the reactor building crane (RBC) analysis considered human errors of commission that may cause an initiating event leading to crane failure. The staff review revealed that this is the dominant failure causing an initiating event. Even though we agree with the reason for excluding detailed hardware failures from the RBC analysis (i.e., the lack of final design details), we believe that this operator error should be identified and included as a risk-significant human action in Table 19.1-70, and also included in Chapter 18.

Section 19.2

This section describes and analyzes the prevention and the mitigation of severe accidents. The section 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.

NuScale evaluated a module's response to the aforementioned spectrum of beyond-design-basis events. Results emphasize the capability of the NuScale design to mitigate severe accidents and phenomena, such as hydrogen generation, high-pressure melt ejection, in-vessel steam explosions, induced steam generator tube failure, and equipment survivability.

NuScale performed severe accident simulations using a modified version of MELCOR to incorporate the NPM unique design features. For severe accidents without containment bypass, results indicate that relocated core materials would be retained within the reactor vessel. NuScale reviewed cases with significant core relocation and selected parameters that they believed would bound heat transfer from the relocated core materials to the reactor vessel. The NuScale evaluation finds that containment integrity would not be challenged in a severe accident.

The staff completed audit calculations using an independently developed input model with a version of the MELCOR code that had been similarly modified. The staff selected three representative severe accident sequences for analysis comparison. For each scenario, no significant differences were found in comparison with NuScale's analysis of severe accident mitigation. The staff also concluded that NuScale had considered an appropriate range of credible core damage scenarios.

Although the MELCOR results provided confidence in the module's response during beyond-design-basis events, the staff observed that the success of in-vessel retention could be impacted by phenomenological uncertainties, such as the potential for stratification of fuels and metals within relocated core materials. Nevertheless, the staff concluded that the potential for large radiation releases from such events would be mitigated by the containment vessel. Were the containment lower head to fail, releases from relocated core debris would be scrubbed sufficiently by the deep reactor pool water to preclude a large release.

The staff has noted that the original design for the bio-shield could allow for the accumulation of hydrogen under the bio-shield during a core-damage event to flammable or combustible concentrations. Such hydrogen combustion may also affect other modules during severe accidents. Consequently, NuScale initiated a bio-shield redesign project to alleviate this concern. The redesign is currently under review by the staff, and it is an open item.

The staff also continues to interact with NuScale regarding the qualification of electrical penetration assemblies for radiation doses associated with core-damage accident scenarios. NuScale recently identified an issue with its evaluation of these assemblies, and the staff is tracking this as an open item as part of its review of the accident source term methodology topical report under revision.

DCA Chapter 21 – Multi-Module Design Considerations

This chapter seeks 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. The applicant discusses the design measures taken to ensure those systems do not introduce significant multi-module risks. The applicant concludes that an accident in one NPM does not result in an accident in another NPM and that no design-basis accidents result from operation or failure of shared systems. The staff has reviewed the information presented by the applicant and documented its findings and conclusions as part of the Chapter 15 review.

NuScale provides information regarding the number of modules supported by various shared systems. However, the documentation is unclear whether all the supported modules must be shut down when a shared system becomes unavailable. We will continue exploring this topic in upcoming meetings with the staff and NuScale.

Summary

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 ITAAC. We have not identified any major issues at this time for Chapter 3, Section 3.9.2 and Chapters 14, 19 and 21. To help identify risk insights in this unique design, there are technical issues in the PRA that merit further consideration.

Members Corradini and Rempe did not participate in Chapter 19, Section 19.2 deliberations.

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 3, Section 3.9.2, 'Dynamic Testing and Analysis of Systems, Components, and Equipment'," April 17, 2019 (ML19102A109).

2. NuScale Power, LLC, "Design Certification Application – Chapter 3, 'Design of Structures, Systems, Components and Equipment, Sections 3.9 to 3.13'," Revision 2, October 30, 2018 (ML18310A323).

3. NuScale Power, LLC, "Design Certification Application – Chapter 3, 'Design of Structures, Systems, Components and Equipment, Appendices 3A to 3C'," Revision 2, October 30, 2018 (ML18310A324).

4. U.S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application – Safety Evaluation with Open Items for Chapter 14, Section 14.2, 'Initial Plant Test Program – Design Certification and New License Applicants'," April 15, 2019 (ML19092A423).

5. U.S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application – Safety Evaluation with Open Items for Chapter 14, Section 14.3, 'Inspections, Tests, Analyses, and Acceptance Criteria'," May 21, 2019 (ML19066A189).

6. NuScale Power, LLC, "Design Certification Application – Chapter 14, 'Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria'," Revision 2, October 30, 2018 (ML18310A336).

7. U.S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application – Safety Evaluation with Open Items for Chapter 19, 'Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors'," April 19, 2019 (ML19073A071).

8. NuScale Power, LLC, "Design Certification Application – Chapter 19, 'Probabilistic Risk Assessment and Severe Accident Evaluation'," Revision 2, October 30, 2018 (ML18310A342).

9. U. S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application – Safety Evaluation With Open Items for Chapter 21, "Multi-Module Design Considerations," April 19, 2019 (ML18220B313).

10. NuScale Power, Design Certification Application, Chapter 21, "Multi-Module Design Considerations," Revision 2, October 30, 2018 (ML18310A344).

11. NuScale Power, LLC, "Technical Report, 'NuScale Power Module Seismic Analysis'," TR-0916-51502, Revision 2, April 2019 (ML19093B850 Publicly Available/ML19094B831 Non-Publicly Available).

12. NuScale Power, LLC, "Technical Report, 'NuScale Power Module Short-Term Transient Analysis'," TR-1016-51669, Revision 0, December 2016 (ML17005A132).

13. NuScale Power, LLC, "Technical Report, 'NuScale Comprehensive Vibration Assessment Program'," TR-0716-50439, Revision 1, January 2018 (ML18022A221 Publicly Available/ML18022A220 Non-Publicly Available).

14. NuScale Power, LLC, "Technical Report, 'NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan'," TR-0918-60894, Revision 0, December 2018 (ML18341A337 Publicly Available/ML18344A069 Non-Publicly Available).

June 19, 2019

SUBJECT: INTERIM LETTER – CHAPTER 3, SECTION 3.9.2, AND CHAPTERS 14, 19 AND 21 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE DESIGN CERTIFICATION APPLICATION REVIEW OF THE NUSCALE SMALL MODULAR REACTOR

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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.

DCA Chapter 5 – Reactor Coolant System and Connecting Systems

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).

May 30, 2019

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

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DATE	5/30/2019	5/30/2019	5/30/2019	5/30/2019	5/30/2019

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PACKAGE: ML19123A276 – Dated May 3, 2019

ML19123A75 - Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards – FY 2018 (27 pages)

ML1923A246 – Memo - Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards – FY 2018

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

May 3, 2019

Mr. Raymond Furstenau, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS - FY 2018

Dear Mr. Furstenau:

Enclosed is our report on the quality assessment of the following research projects:

- NUREG-2218, "An International Phenomena Identification and Ranking Table (PIRT) Expert Elicitation Exercise for High Energy Arcing Faults (HEAFs)"
- NUREG/CR-7237, "Correlation of Seismic Performance in Similar SSCs (Structures, Systems, and Components)"

These projects were selected from a list of projects provided by the Office of Nuclear Regulatory Research. We found each to be a satisfactory and professional product that satisfies its research objectives.

We anticipate receiving a list of candidate projects for quality assessment in FY 2019 prior to our June 2019 meeting.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

Enclosure:
As stated

May 3, 2019

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS - FY 2018

Package No: ML19123A276
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DATE	5/2/2019	5/2/2019	5/3/2019	5/3/2019	5/3/2019

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

April 17, 2019

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

SUBJECT: INTERIM LETTER: CHAPTERS 9, 10, 11, 12 AND 16 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 662nd meeting of the Advisory Committee on Reactor Safeguards, April 4-5, 2019, we met with representatives of NuScale Power, LLC (NuScale) and the NRC staff to review Chapter 9, "Auxiliary Systems," Chapter 10, "Steam and Power Conversion Systems," Chapter 11, "Radioactive Waste Management," Chapter 12, "Radiation Protection," and Chapter 16, "Technical Specifications," 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 March 20-21, 2019. During these meetings, we had the benefit of discussions with NuScale and the staff. We also had the benefit of the referenced documents.

CONCLUSION AND RECOMMENDATION

1. There are potentially risk-significant items in the NuScale design that are not yet fully developed. For these items, requirements should be included in the DCA to ensure that the licensed NuScale plant will perform as credited.

2. We have not identified any additional major issues at this time for Chapters 9, 10, 11, 12 and 16.

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 (COL) applicant must provide. Our review is being conducted on a chapter-by-chapter basis to identify issues that may merit further consideration by the staff. This process aids in the resolution of concerns and facilitates timely completion of the design

DCA Chapter 10 – Steam and Power Conversion Systems

The steam and power conversion system removes thermal energy from the reactor coolant system and transfers it to the main turbine generator. The main elements of the steam and power conversion system include the main steam, turbine generator, turbine bypass, main condensers, circulating water, condensate polishing, feedwater treatment, condensate and feedwater, and auxiliary boiler systems. The power conversion system is not safety related and is not required for safe shutdown. However, the main steam and main feedwater systems have piping that penetrate the containment and components that directly interface with safety-related structures, systems, and components (SSCs). The failure of these components can have an adverse impact on plant safety and the plant's ability to achieve a safe shutdown.

In particular, the main turbine-generator is oriented such that some safety-related equipment is in the turbine low-trajectory hazard zone of potential missiles generated by a turbine-generator catastrophic failure. NuScale has chosen to protect these vulnerable components using strong physical barriers. The analysis supporting the acceptability of these barriers is described in Chapter 3, "Design of Structures, Systems, Components and Equipment," of the DCA. We defer judgment on the acceptability of this approach until our review of Chapter 3.

DCA Chapter 11 – Radioactive Waste Management

The NuScale radioactive waste management system consists of the liquid radioactive waste system, gaseous radioactive waste system, solid radioactive waste system, and process and effluent radiation monitoring instrumentation and sampling system. These systems are designed for normal operations, including refueling outages, routine maintenance, and anticipated operational occurrences. As operational events, anticipated operational occurrences include unplanned releases of radioactive materials associated with fuel failures, equipment failures, operator errors, and administrative errors, with radiological consequences that are not considered accident conditions.

NuScale has proposed to use EPRI historical data as a conservative bound for the realistic failed fuel fraction (RFFF). Staff reviewed the information for calculating the revised RFFF and found it acceptable for source terms during normal operations. Staff also confirmed that Revision 1 of NuScale TR-1116-52065-P and Revision 2 of FSAR Tier 2, Sec. 11.1, included this revised RFFF and corresponding source terms during normal operations. The design basis failed fuel fraction is assumed to be an order of magnitude greater than the RFFF. The staff is still reviewing the use of the design basis failed fuel fraction as a source term for radiation shielding, ventilation systems, and radiation zoning. This is being tracked as an open item.

With the exception of the open items, we agree with the staff's conclusions that the NuScale design-basis source term, the realistic source term, and the radioactive waste management systems comply with the regulatory requirements.

DCA Chapter 12 – Radiation Protection

This chapter provides information on facility and equipment design and programs used to meet the radiation protection requirements in 10 CFR Part 20, "Standards for Protection against Radiation"; 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

SUMMARY

There are potentially risk-significant items in the NuScale design that are not yet fully developed. For these items, requirements should be included in the DCA to ensure that the licensed NuScale plant will perform as credited. We have not identified any additional major issues at this time for Chapters 9, 10, 11, 12 and 16.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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2. NuScale Power, Design Certification Application, Chapter 9, "Auxiliary Systems," Revision 1, March 15, 2018 (ML18086A180).
3. U.S. Nuclear Regulatory Commission, SECY 94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994 (ML003708068).
4. U.S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application – Safety Evaluation With Open Items for Chapter 10, 'Steam and Power Conversion System'," January 23, 2019 (ML18257A276).
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6. U.S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application – Safety Evaluation With Open Items for Chapter 11, 'Radioactive Waste Management'," January 17, 2019 (ML19015A051).
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April 17, 2019

SUBJECT: INTERIM LETTER: CHAPTERS 9, 10, 11, 12 AND 16 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE DESIGN CERTIFICATION APPLICATION REVIEW OF THE NUSCALE SMALL MODULAR REACTOR

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

April 17, 2019

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

SUBJECT: ACRS REVIEW OF APPLICATIONS FOR OPERATION IN THE EXPANDED POWER TO FLOW DOMAIN

Dear Ms. Doane:

During the 662nd meeting of the Advisory Committee on Reactor Safeguards, April 4-5, 2019, we evaluated our involvement in reviews of the License Amendment Request (LAR) in the expanded power to flow domain, such as the Maximum Extended Load Limit Analysis Plus (MELLLA+) operating domain. Our discussion was informed by materials presented at the March 19, 2019, meeting of the ACRS Power Uprates Subcommittee in which representatives from the staff and their consultants reported their progress on a review of the license amendment request (LAR) by Tennessee Valley Authority (TVA) for Units 1, 2, and 3 of the Browns Ferry Power Plant (Browns Ferry) and on other activities affecting operation in the expanded power to flow domain. Representatives from TVA also provided input regarding their Browns Ferry LAR during this Subcommittee meeting.

The staff has established a mature process for completing reviews on this topic and applied this process to a range of BWR designs. In more complex cases, such as ones in which a licensee combines methods developed by multiple vendors, the staff completes TRACE confirmatory calculations to increase confidence that appropriate uncertainties are considered by the licensee.

Hence, we conclude:

- No additional ACRS review of the TVA MELLLA+ LAR for Browns Ferry is required unless there are substantive changes in the status that the staff provided on March 19, 2019.

- ACRS participation in future reviews of this topic is not required unless a new LAR involves substantive differences from the plant designs and conditions that have been approved.

We request that the staff continue to inform us of future activities relevant to this topic that might impact the above conclusions.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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3. Tennessee Valley Authority, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus – Supplement 2, Operator Training Results," July 23, 2018 (ADAMS Accession No. ML18205A498).

4. Tennessee Valley Authority, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus – Supplement 3, Responses to Requests for Additional Information," December 14, 2018 (ADAMS Accession No. ML18348B156).

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Supplement 7, Responses to Requests for Additional Information," January 25, 2019 (ADAMS Accession No. Ml19025A204 and ML19025A205 (proprietary)).

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10. U.S. Nuclear Regulatory Commission, "Browns Ferry Maximum Extended Load Line Limit Plus (MELLLA+) Safety Evaluation Overview" (Official Use Only), February, 2019.

April 17, 2019

SUBJECT: ACRS REVIEW OF APPLICATIONS FOR OPERATION IN THE EXPANDED POWER TO FLOW DOMAIN

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

March 21, 2019

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

SUBJECT: INTERIM LETTER: CHAPTERS 13 AND 18 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 661st meeting of the Advisory Committee on Reactor Safeguards, March 7-8, 2019, we met with representatives of NuScale Power, LLC (NuScale) and the NRC staff to review Chapter 13, "Conduct of Operations," and Chapter 18, "Human Factors Engineering," 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 January 23, 2019. During this meeting, we had the benefit of discussions with NuScale and the staff. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATIONS

1. Operator training drills should include scenarios where computer displays provide misleading or incomplete information to ensure operators maintain alternate diagnostic approaches.
2. The human factors engineering program review needs to be coordinated with the review of reactor building crane design features and operations in subsequent design certification chapters in order to minimize any hazards from heavy load lifts, including module movement.
3. We have not identified any additional major issues at this time for Chapters 13 and 18.

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 (COL) applicant must provide.

Our review is being conducted on a chapter-by-chapter basis to identify issues that may merit further consideration by the staff. This process aids in the resolution of concerns and facilitates

timely completion of the design certification application review. Our review addresses the staff's SER and DCA Chapter 13, Revision 1 and Chapter 18, Revision 1 along with supplemental material, including NuScale responses to staff requests for additional information.

DISCUSSION

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

DCA Chapter 13 – Conduct of Operations

The SER on Chapter 13 summarizes the requirements for the COL applicant in the areas of:

- The management and technical support organization,
- Description and schedule of initial personnel training and qualification,
- Design features, facilities and equipment used to support emergency response functions,
- Site-specific information for operational programs,
- Administrative and operating procedures used by the operation organization to ensure activities are conducted in a safe manner

The DCA Chapter 13 topics on the physical security plan, cyber security plan and fitness for duty were not reviewed in this SE because they contain security related information.

There is significant overlap between Sections 13.4, "Operational Programs," 13.5, "Plant Procedures," and all of Chapter 18. The one significant open item in Chapter 13 applies equally well to Chapter 18. It is related to the staff review of NuScale Generic Technical Guidelines (GTGs) focused on (1) the three critical safety functions (CSFs) defined by NuScale, (2) the methodology used to identify key operator actions, and (3) the CSF flowchart logic and operator actions necessary to assess and maintain these functions. The applicant has submitted responses to the staff's requests for additional information. At present, the staff is unable to conclude the NuScale GTGs are acceptable for use as a basis for the development of COL applicant plant specific technical guidelines. NuScale has performed Integrated System Validation (ISV) testing, which will provide needed input for the staff evaluation, as well as any necessary changes to the GTGs and the associated post-accident monitoring variables. The ISV activity and report will be completed as part of the DCA and are being followed as an Open Item.

The staff tentatively concluded that passive reactivity management during an anticipated transient without scram can be credited. However, this is still under review as parts of Chapters 15 and 19. Thus, until these reviews are completed it is premature to draw this conclusion here.

NuScale plant operations rely on computer assistance, including summary displays of plant status. That assistance will be helpful to the operators and protects against some of the most common human errors. However, it may also result in an over reliance on computer aids. NuScale noted that, as part of operator training, black-screen scenarios are conducted with primary computer failures and operators must rely on direct diagnosis of plant instrumentation data. We note that partial computer failures may be more limiting than complete failures. Operator training drills should include scenarios where computer displays provide misleading or incomplete information to ensure operators maintain alternate diagnostic

approaches. Operators should not rely exclusively on high-level computerized information but check and verify available redundant plant information for appropriate response.

DCA Chapter 18 – Human Factors Engineering

This chapter describes the human factors engineering (HFE) program for the NuScale plant. The HFE program takes advantage of state-of-the art technology and incorporates accepted HFE standards and guidelines, including the applicable staff guidance (NUREG-0711). The staff reviewed the HFE program under four general activities described in NUREG-0711:

- Planning & analysis - HFE program management, operating experience review, functional requirements and function allocation, task analysis, staffing and qualifications, and treatment of 'important human actions'
- Design - human-system interface design, procedure development, and training program development
- Human factors verification and validation
- Operational implementation - implementation and operation, and design implementation

In addition to developing a well-defined HFE program, NuScale has gone further at the DCA stage, completing many of the tasks included in their program. The results of this comprehensive work are reported in technical reports cited in Chapter 18 and in citations in those reports. Some have been included by reference in Chapter 1.

The NuScale simulator control panel layout and proposed operational practices evolved from testing with operators. The staffing and qualifications program is based on actual tests for a variety of scenarios. It is anchored to operating experience reviews, functional requirements analysis and function allocation, task analysis, and development of human-machine interface, for which there are technical reports.

The SE has 23 open items, most of which should be resolved when the staff reviews the results of NuScale's verification and validation program, including the ISV. All of the open items are confirmatory in nature. Closure of some items will require completion of the SERs for Chapters 7, 15 and 19.

A unique feature of the NuScale design involves heavy load lift and movement adjacent to operating reactors. The 12 module NuScale plant will have module movement for refueling every two months. Each refueling requires a NuScale module to be removed and relocated to the refueling station. When refueled, the module is returned to its original location. Such heavy load lift operations will be frequent, involving constant activity adjacent to operating reactors.

The HFE issues related to heavy load lifting and module movement deserve attention. The staff appropriately identified the notion of risk relative to the reactor-building crane and its operation. NuScale noted that these HFE issues pertaining to module movement will be addressed by the crane vendor. We emphasize that the applicant is responsible for this HFE analysis and the staff is expected to review it at the DCA or COL stage. In addition, NuScale design features not currently included and detailed for module movement could be incorporated to reduce the probability that a load drop would damage an operating module or the pool wall. The HFE program review needs to be coordinated with a review of these potential design features in

subsequent DCA chapters to minimize any hazards from heavy load lifts that include module movement. This also includes HFE review considering incremental installation of additional modules while some are already operating.

SUMMARY

We have identified some items that need to be resolved. However, we have not identified any additional major issues at this time for Chapters 13 and 18.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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5. NuScale Power, Design Certification Application, Chapter 18, "Human Factors Engineering," Revision 1, March 15, 2018, (ML18086A065).

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March 21, 2019

SUBJECT: INTERIM LETTER: CHAPTERS 13 AND 18 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE DESIGN CERTIFICATION APPLICATION REVIEW OF THE NUSCALE SMALL MODULAR REACTOR

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

March 19, 2019

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

SUBJECT: DRAFT SECY PAPER AND GUIDANCE DOCUMENTS TO IMPLEMENT A TECHNOLOGY-INCLUSIVE, RISK-INFORMED, AND PERFORMANCE-BASED APPROACH TO INFORM THE CONTENT OF APPLICATIONS FOR LICENSES, CERTIFICATIONS, AND APPROVALS FOR NON-LIGHT-WATER REACTORS

Dear Chairman Svinicki:

During the 660th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 2019, we reviewed the draft SECY paper entitled, "Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," and the associated draft Regulatory Guide DG-1353 and Nuclear Energy Institute (NEI) guidance document NEI 18-04. Our Future Plant Designs Subcommittee also reviewed this matter during meetings on June 22 and October 30, 2018. During these meetings we had the benefit of discussions with representatives of the NRC staff and of the nuclear power industry. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATION

1. The draft SECY paper proposes the next evolution of a licensing approach that has been developed over the past thirty years.

2. The paper proposes an approach to accomplish three objectives: to select licensing basis events; to classify structures, systems, and components; and to assess the adequacy of defense-in-depth for new designs. The approach has matured to the point of being ready for application.

3. We recommend that the Commission adopt the approach proposed by the staff for a technology-inclusive, risk-informed, and performance-based methodology for informing the licensing basis and content of applications for non-light-water reactors.

4. The guidance proposed in DG-1353 is adequate to support implementation of the approach described in the SECY paper, with the exception that guidance for developing mechanistic source terms should be expanded.

5. DG-1353 should be finalized and published for comment.

BACKGROUND

The Commission policy statement regarding advanced reactors, issued on October 14, 2008, outlined the Commission expectation that advanced reactors provide at least the same degree of protection of the environment and public health and safety, and the common defense and security as is required for current generation light-water reactors. Additionally, the Commission expected enhanced margins of safety and use of simplified, inherent, passive, or other innovative means to accomplish these safety and security functions. The proposed approach neither exempts any reactor design from existing regulations, nor does it address all regulations applicable to nuclear power plants.

As the staff prepares to review and regulate the new generation of non-light-water reactors (non-LWRs), the NRC developed a vision and strategy to assure NRC readiness to efficiently and effectively conduct its mission for these technologies, including fuel cycles and waste forms. "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Mission Readiness," published in December 2016, is the overarching document that describes the objectives, strategies, and contributing activities necessary to achieve non-LWR mission readiness.

The project was organized into two phases. Phase 1 was the conceptual planning phase that was completed in December 2016, with the issuance of the final non-LWR Vision and Strategy Document. Phase 2 includes detailed work planning efforts and task execution, including the development of implementation action plans (IAPs). Six parallel strategies were identified for achieving non-LWR licensing readiness. In our report of March 21, 2017, we recommended that the highest priority be given to implementing action plan Strategy 3, a flexible regulatory review process, and Strategy 5, technology-inclusive policy issues. We wrote several reports as that work progressed including those on the staff's guidance for developing principal design criteria, functional containment performance criteria, and emergency preparedness for small modular reactors and other new technologies. The current SECY paper is the culmination of the staff work.

Phase 2 is broken down into three periods: near-term (0-5 years), mid-term (5-10 years), and long-term (greater than 10 years). The near-term actions have been further developed. The purpose of the IAPs is to identify specific, actionable tasks that, once completed, will lead to accomplishment of the NRC's non-LWR vision and strategy objectives: enhance technical readiness, optimize regulatory readiness, and optimize communications.

Among the six strategies in the near-term IAPs, Strategy 3 is "Develop guidance for a flexible non-LWR regulatory review process within the bounds of existing regulations, including the use of conceptual design reviews and staged-review processes." Strategy 3 has activities in support of the following:

1. Establish criteria, as necessary, to reach a safety, security, or environmental finding for non-LWR technologies;

2. Determine appropriate licensing bases and accident sets for non-LWR technologies;

3. Identify and resolve gaps in current regulatory framework associated with non-LWR reactors and the associated fuel cycle;

4. Develop a regulatory review "roadmap" reflecting design development lifecycle and appropriate interactions, including potential research and test reactor interactions;

5. Update prototype reactor guidance;

6. Engage on technology- or design-specific licensing project plans and develop regulatory approaches commensurate with the risks posed by the technology;

7. Support longer-term efforts to develop, as needed, a new non-LWR regulatory framework that is risk-informed and performance-based, and that feature staff review efforts commensurate with the demonstrated safety performance of the non-LWR nuclear power plant design being considered.

The technology-inclusive, risk-informed, and performance-based approach has evolved over the past thirty years starting with the U.S. Department of Energy (DOE)/General Atomics modular high-temperature gas-cooled reactor application in 1986 and refined during the NRC's development of NUREG-1860, and DOE's Next Generation Nuclear Plant white papers. These were addressed in our letters of October 13, 1988, September 26, 2007, and May 15, 2013.

The current *Licensing Modernization Project* is an effort led by Southern Company, with others from industry contributing, and cost-shared by DOE. It has features that support activities number (2) and (7), and depending on the outcomes, perhaps contribute to activity number (1). The project builds on best practices, as well as previous activities through DOE and industry-sponsored non-LWR licensing initiatives. This effort was initiated with the submittal of four working draft documents, which were commented on by NRC staff. Those four documents, with NRC comments addressed, were compiled into NEI 18-04.

DISCUSSION

The staff has prepared a draft SECY paper and guidance document (DG-1353) to implement a technology-inclusive, risk-informed, and performance-based approach to inform the content of applications for licenses, certifications, and approvals for non-light-water reactors. The SECY paper seeks Commission approval to adopt that methodology. The draft regulatory guide endorses, with clarifications, the methodology documented in NEI 18-04. The methodology would be used by licensees to select licensing basis events; to classify structures, systems, and components; and to assess the adequacy of defense-in-depth for new designs. This is an iterative approach, one beginning in the design stage and continuing through operations.

The draft SECY paper and DG-1353 are an integrated set, with the paper describing the new approach and the DG describing how the approach can be applied. In the SECY paper, the staff argues convincingly that the proposed methodology is consistent with 2008 Commission policy statement on advanced reactors, and the Staff Requirements Memoranda for SECY-03-0047 and SECY-15-0168.

Over the past thirty years, we have supported careful development of the evolving approach described in the SECY paper, given that the gaps in the methodology would be addressed and undefined details would be developed. The paper emphasizes three technical areas.

For the identification of licensing basis events (LBEs), the recommended approach is essentially the same as described in NUREG-1860 and the Next Generation Nuclear Plant white papers. The licensing basis events are defined by scenarios developed in a probabilistic risk

assessment (PRA) that meets NRC standards. The PRA must be based on design-specific required safety functions. LBEs are tested against frequency-consequence (F-C) target goals given in NEI 18-04. Scenarios that do not meet these goals must be improved, by developing design, procedural, or administrative changes to lower the scenario frequency or consequences. Comparison of LBEs against the F-C target must be done in a way that prevents the analyst from arbitrarily splitting one scenario into many more scenarios with much lower frequencies. This is generally accomplished by combining scenarios into LBE families, each having similar initiating event, challenge to a PRA safety function, plant response, end state, and mechanistic source term. The total integrated plant risk must also meet separate integrated risk goals.

The LBEs include anticipated operational occurrences, design basis events, and beyond design basis events, all now defined unambiguously by the PRA frequency results. The staff has set a lower bound on beyond design basis events of 5×10^{-7} per year. Design basis accidents are derived from the design basis events, based on the capabilities and reliabilities of safety-related structures, systems, and components (SSCs) needed to mitigate and prevent the event sequences. They are used to set design criteria and performance objectives for the design of safety-related SSCs.

For classifying SSCs, the paper extends and makes operational the concepts that were expressed in NUREG-1860 and the Next Generation Nuclear Plant white papers. Essentially, SSCs are selected from important risk contributors in the PRA, with special treatment assigned based on importance to risk. The safety classification process and the corresponding special treatments serve to control the frequencies and consequences of the LBEs in relation to F-C target goals and ensure that cumulative risk metrics are not exceeded. We previously addressed these concepts in reports of September 26, 2007, and April 26, 2012, and find the approach logically sound, based on safety importance, and not bound by historical practice.

For defense-in-depth, the paper provides an operational structure for evaluation. It bridges the gap between frameworks as described in NUREG-1860, NUREG/KM-0009, and NUREG-2150, and viable regulatory actions. The paper describes a defense-in-depth approach that includes probabilistic and deterministic assessment techniques using a combination of plant capabilities and programmatic controls. As part of the evaluation, each LBE is evaluated to confirm that risk targets are met without exclusive reliance on a single element of design, single program, or single defense-in-depth attribute.

NEI 18-04 explains that one of the primary motivations of employing defense-in-depth attributes is to address uncertainties, including those that are reflected in the PRA estimates of frequency and consequence, as well as other uncertainties which are not sufficiently characterized for uncertainty quantification and are not amenable to sensitivity analyses. The designer is to convene an Integrated Decision Panel to make the kinds of judgments involving quantitative and qualitative factors akin to the integrated decision process in Regulatory Guide 1.174. The panel supports the overall design effort including selection of LBEs and classification of SSCs, and conducts the defense-in-depth adequacy evaluation for the design. This process has not been fully deployed in any of the table top exercises. Therefore, although the approach appears to be well conceived, we reserve judgment on how the guidance prepares the applicant to effectively carry out the integrated decision process.

DG-1353 provides guidance on using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and the content of applications for non-LWRs. It endorses, with clarifications, the principles and methodology in NEI 18-04, as one acceptable method for determining the appropriate scope and level of detail for parts of applications for

licenses, certifications, and approvals for non-LWRs. The guidance in DG-1353 and NEI 18-04 are adequate for an applicant to successfully prepare an acceptable application. One area that remains vague in both documents is how an applicant should develop mechanistic source terms for scenarios to be used in the PRA, which is a design-specific process. Developing a mechanistic source term for every scenario in the PRA is no easy task. It involves complex physics and chemical phenomena, including the evolution and transport of aerosols. Applicants need to know the level of detailed analysis and experimentation that will be required.

SUMMARY

The approach presented in the SECY paper is ready for use to select LBEs; to classify SSCs; and to assess the adequacy of defense-in-depth for new designs. We look forward to following the staff's efforts, as described in the SECY paper and DG-1353. We expect to see applications using this methodology and the staff's approach for review of such submittals.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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1. U.S. Nuclear Regulatory Commission, SECY Paper, "Technology-Inclusive, Risk-Informed, and Performance-Based Approach To Inform The Content of Applications For Licenses, Certifications, and Approvals For Non-Light-Water Reactors," Draft, September 28, 2018 (ML18270A334).

2. U.S. Nuclear Regulatory Commission, Regulatory Guide DG-1353, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," Draft, September 28, 2018 (ML18271A164).

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5. U.S. Nuclear Regulatory Commission, "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness," December 21, 2016 (ML16356A670).

6. Advisory Committee on Reactor Safeguards, "NRC Non-Light Water Reactor Vision & Strategy – Near-Term Implementation Action Plans and Advanced Reactor Design Criteria," March 21, 2017 (ML17079A100).

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9. Idaho National Laboratory, "Contract No. DE-AC07-05ID14517 – Next Generation Nuclear Plant Licensing White Paper Submittal – Fuel Qualification – NRC Project #0748," July 21, 2010 (ML102040261).

10. Idaho National Laboratory, "Contract No. DE-AC07-05ID14517 – Next Generation Nuclear Plant Licensing White Paper Submittal – Mechanistic Source Terms – NRC Project #0748," July 21, 2010 (ML102040260).

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13. Idaho National Laboratory, "Contract No. DE-AC07-05ID14517 – Next Generation Nuclear Plant Licensing White Paper Submittal – Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification – NRC Project #0748," September 21, 2010 (ML102660144).

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March 19, 2019

SUBJECT: DRAFT SECY PAPER AND GUIDANCE DOCUMENTS TO IMPLEMENT A TECHNOLOGY-INCLUSIVE, RISK-INFORMED, AND PERFORMANCE-BASED APPROACH TO INFORM THE CONTENT OF APPLICATIONS FOR LICENSES, CERTIFICATIONS, AND APPROVALS FOR NON-LIGHT-WATER REACTORS

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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WASHINGTON, DC 20555 - 0001

March 7, 2019

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

SUBJECT: EDO RESPONSE TO ACRS LETTER OF SEPTEMBER 26, 2018 ON CHAPTERS 7 AND 8 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE NUSCALE SMALL MODULAR REACTOR

Dear Ms. Doane:

During the 660th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 2019, we reviewed the NRC staff's October 30, 2018 letter regarding disposition of the conclusion and recommendations in our NuScale letter of September 26, 2018, on Chapters 7 and 8 of the staff's safety evaluation report with open items related to the certification of the NuScale small modular reactor.

Our Recommendation 2 stated:

> The staff should ensure that the unidirectional communication interfaces labeled on Figure 7.0-1 in Chapter 7 of NuScale's design certification application as "PCS Unidirectional Data Diode" and "MCS Unidirectional Data Diode" are one-way, hardware-based devices that neither use nor are configured by software to demonstrate complete isolation from external communications.

On October 24, 2018, NuScale submitted mark-ups of Chapter 7 of the Design Certification Application (DCA) that will be included in a future revision as documented in NuScale letter LO-1018-62193. This provided additional clarifications to the Chapter 7 sections in question; specifically, the mark-ups clarify the design of one-way deterministic isolation devices. The clarifications indicate that the one-way deterministic isolation device between the module control system (MCS) and plant control system (PCS) to the plant network (Figure 7.0-1) transmits network traffic from the MCS and PCS to the plant network in one direction only, which is enforced in the hardware design, not software. No software configuration or misconfiguration will cause the boundary device to reverse the direction of data flow.

Our Recommendation 2 in our September letter will be resolved when these proposed changes to the DCA are incorporated into a future revision of Chapter 7, as the NuScale October 24, 2018 letter indicates.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Chapter 7, 'Instrumentation and Controls,' and Chapter 8, 'Electric Power' of the U.S. Nuclear Regulatory Commission Staff's Safety Evaluation Report with Open Items Related to the Certification of the NuScale Power, LLC Small Modular Reactor," October 30, 2018 (ML18275A389).

2. Advisory Committee on Reactor Safeguards, "Interim Letter: Chapters 7 and 8 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the NuScale Small Modular Reactor," September 26, 2018 (ML18270A374).

3. NuScale Power, LO-1018-62193, "NuScale Power, LLC Submittal of Changes to Final Safety Analysis Report, Section 7.0, 'Instrumentation and Controls – Introduction and Overview,' and Section 7.2, 'System Features'," October 24, 2018 (ML18298A222).

March 7, 2019

SUBJECT: EDO RESPONSE TO ACRS LETTER OF SEPTEMBER 26, 2018 ON CHAPTERS 7 AND 8 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE NUSCALE SMALL MODULAR REACTOR

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

February 28, 2019

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

SUBJECT: NON-POWER PRODUCTION OR UTILIZATION FACILITIES PROPOSED LICENSE RENEWAL RULEMAKING

Dear Chairman Svinicki:

During the 660th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 2019, we completed our review of the draft final Non-Power Production or Utilization Facilities Proposed License Renewal rulemaking. We had previously reviewed this topic during our 632nd meeting of the Committee and issued a letter report, dated March 15, 2016, on the proposed rule. The Research and Test Reactors Subcommittee also reviewed these matters at its meetings on February 3, 2016, and January 23, 2019. During our review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the referenced documents.

RECOMMENDATION

The staff should proceed with this rulemaking for license renewal of Non-Power Production or Utilization Facilities.

BACKGROUND

The Atomic Energy Act includes considerations for licensing research reactors and testing facilities, as well as considerations for licensing commercial nuclear power reactors. The Atomic Energy Act accords to research reactors and testing facilities special status and specifies that these reactors be subject to minimal regulation consistent with adequate protection of the public health and safety. The current regulatory process for these reactors permits renewable licenses of twenty-year duration. For a variety of reasons, a backlog in the processing of the license renewals has developed. The staff has undertaken a revision of the regulations to avoid recurrence of such a backlog and to improve the safety documentation for the research reactors and testing facilities.

Our original letter report dated March 15, 2016, provides an initial review of the proposed rulemaking prior to the public comment phase of the review process. In the draft proposed rule, the staff proposed nine changes to the regulations:

- Define "Non-power Production or Utilization Facilities" (NPUFs)
- Eliminate license terms for research reactors
- Consolidate license renewal requirements for testing facilities and medical isotope production facilities
- Require NPUF licensees to submit updated final safety analysis reports every five years
- Amend the timely license renewal provision under 10 CFR 2.109, "Effect of timely renewal application"
- Provide an accident dose criterion of 1 rem (0.01 Sievert [Sv]) for NPUFs other than test reactors
- Extend applicability of 10 CFR 50.59, "Changes, tests, and experiments," to NPUFs regardless of the decommissioning status
- Clarify existing environmental reporting requirements
- Eliminate NPUF financial qualification information requirements for license renewal

We commented on the first seven of these proposed changes in our original letter report and deferred on the final two proposed changes as they are outside our charter and expertise. In that same letter report, we recommended that the staff should proceed with this proposed rulemaking for licensing of NPUFs. The staff has now received public and stakeholder comments on the proposed rule and is recommending material revisions to some of the originally proposed changes. This current letter report documents our review and recommendation on the staff changes to the draft final rule that now addresses the public and stakeholder comments. For completeness, we also discuss significant aspects of the proposed rule from our original letter report.

DISCUSSION

The staff initially proposed to define NPUF as a term that encompasses:

- Research reactors with power of 10 MWt or less if there are no notable safety considerations, and 1 MWt or less if there are notable safety considerations[1]
- Testing facilities with power greater than 10 MWt, or greater than 1 MWt if there are notable safety considerations
- Commercial medical radioisotope irradiation and production facilities

Research reactors typically operate at low pressures and have low inventories of radionuclides. On the other hand, research reactors often have small exclusion area boundaries, and are often located in high population areas. The accident dose criterion found in 10 CFR Part 100 (25 rem (0.25 Sv) total effective dose equivalent (TEDE)) appears inappropriately large for these facilities. The radiation dose limit for individual

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[1]The term, "notable safety considerations," refers to circulating loops through the reactor core used for fuel experiments, liquid fuel loading, and experimental volumes with cross-sections larger than 16 in^2 (103 cm^2) in the core.

members of the public (0.1 rem (0.001 Sv) TEDE) established by 10 CFR Part 20 appears unduly restrictive as an accident dose criterion. The staff proposed to adopt the 1 rem (0.01 Sv) TEDE Protective Action Guide defined by the Environmental Protection Agency as an accident dose criterion for research reactors. Testing facilities will remain subject to the 10 CFR Part 100 accident dose criterion.

The staff received public comments on the definitions of "testing facility" and "research reactor." The staff agreed that "testing facility" and "research reactor" as defined by a threshold of 10 MWt was arbitrary and overly restrictive. Instead, a risk-based approach to regulation of a testing facility would be more appropriate. The staff concluded that the accident dose criterion of 1 rem (0.01 Sv) TEDE be used in the draft final rule to define the threshold between a research reactor and testing facility. The technical basis associated with the 10 MWt threshold, while generally based on safety significance, is not documented. These prescriptive power thresholds do not take into account the safety features that are engineered into the facility design, and therefore, do not accurately represent the risk associated with a particular facility. For these reasons, the use of a calculated accident dose is a more risk-informed, performance-based approach than the power level of the reactor to distinguish between types of NPUFs. In the draft final rule, the staff revises the definitions of "testing facility" and "research reactor" to reflect this approach. We concur.

Upon reviewing public comments, the staff concluded that the draft proposed rule was too broad for defining production facilities that are NPUFs. Previously, the NPUF definition excluded fuel reprocessing plants, but did not address the additionally required exclusion for a production facility designed or used primarily for the formation of plutonium or uranium-233, or designed for the separation of the isotopes of plutonium. In the draft final rule, the staff revises the definition to exclude all production facilities as defined under paragraphs (1) and (2) of the definition of "production facility" in 10 CFR 50.2. We concur.

The staff proposed that research reactors be given licenses that do not expire, once they have renewed their licenses following the guidance provided in NUREG-1537. The NRC will continue to monitor and inspect these reactor facilities as they have in the past and the licensees will remain obligated to report any deviations from technical specifications. Licensees will be required to provide to the NRC, updates to their final safety analysis report every five years. Submission of updates to the final safety analysis report should assure adequate attention to configuration control of the research reactors and that licensees have adequate familiarity with the licensing bases of their facilities. This requirement for systematic and periodic reexamination provides added confidence that changes which may affect safety are identified and managed throughout the life of the facility.

The staff proposed that applications for license renewal be submitted at least two years prior to expiration of the license. This proposal is being made to allow sufficient time for the staff to review the renewal application without complications of license expiration during the review period.

In the draft proposed rule, a renewed license would have been effective 30 days after issuance. The staff has modified the draft proposed rule language, after public comments, to allow for potentially greater flexibility to the licensee regarding the date of issuance. The applicant for the renewed license can propose a schedule for implementation of the renewed license to the extent that additional time is needed to make any necessary and conforming changes to the facility processes and procedures required by the applicable conditions of the renewed license. We concur.

Current wording in the regulations makes 10 CFR 50.59 no longer applicable to NPUFs that have ceased operation and have returned their fuel to the Department of Energy. This has mandated that NRC consider license amendments and add license conditions for decommissioning facilities that are essentially identical to the requirements of 10 CFR 50.59. The staff proposed changes to the regulations that eliminate this additional administrative burden and make 10 CFR 50.59 applicable to NPUFs regardless of decommissioning status.

Licensing terms for testing facilities and for commercial or industrial facilities (e.g., medical radioisotope production facilities) will continue much as they have in the past. The staff proposed to consolidate the regulatory requirements for renewal of these licenses in a new section of the *Code of Federal Regulations*, 10 CFR 50.135.

CONCLUSION

The staff has developed a practical revision to the licensing process for NPUFs that is well conceived and should serve to reduce administrative challenges that have arisen in the past while preserving the adequate protection of the health and safety of the public.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

REFERENCES

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2. U.S. Nuclear Regulatory Commission, Draft SECY Paper, "Proposed Rulemaking: Non-Power Production or Utilization Facility License Renewal (RIN 3150-A196)," February 25, 2016 (ML16055A134).

3. U.S. Environmental Protection Agency, EPA 400/R-17/001, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," January 2017.

4. Advisory Committee on Reactor Safeguards, "Non-Power Production or Utilization Facilities Proposed License Renewal Rulemaking," March 15, 2016 (ML16075A306).

5. U.S. Nuclear Regulatory Commission, NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," February 1996 (ML12251A353).

February 28, 2019

SUBJECT: NON-POWER PRODUCTION OR UTILIZATION FACILITIES PROPOSED LICENSE RENEWAL RULEMAKING

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UNITED STATES
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WASHINGTON, DC 20555 - 0001

February 22, 2019

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

SUBJECT: SAFETY EVALUATION FOR ANP-10332P, REVISION 0, AURORA-B: AN EVALUATION MODEL FOR BOILING WATER REACTORS; APPLICATION TO LOSS-OF-COOLANT ACCIDENT SCENARIOS

Dear Ms. Doane:

During the 660th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 2019, we completed our review of report ANP-10332P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios," and the associated NRC staff draft safety evaluation (SE). Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter on December 18, 2018. During these reviews, we benefitted from discussions with representatives of the staff, their contractors, and Framatome. We also benefitted from the referenced documents.

CONCLUSION AND RECOMMENDATION

1. The AURORA-B loss-of-coolant accident evaluation model provides an acceptable methodology to estimate safety margins for boiling-water reactors during loss-of-coolant events in accordance with Appendix K to 10 CFR Part 50.

2. The staff's safety evaluation provides a comprehensive evaluation of this methodology and imposes limitations and conditions to ensure its appropriate application. It should be published.

BACKGROUND

AURORA-B is a multi-physics, multi-code package developed for predicting the dynamic response of boiling-water reactors (BWRs) during a variety of transient and accident scenarios. We completed our review of applications of AURORA-B for anticipated operational occurrences in October 2017 and control rod drop accidents in March 2018. Together, these applications cover most of the transient and accident events described in Chapter 15 of the NRC's Standard Review Plan (NUREG-0800). The only exceptions are stability events, which will continue to use the RAMONA-5FA methodology, and the latter stages of anticipated transient without scram scenarios, which are not normally evaluated for each fuel reload.

The latest topical report, ANP-10332P, extends AURORA-B applicability to loss-of-coolant accident (LOCA) analysis to demonstrate compliance with the criteria in 10 CFR 50.46. Framatome has chosen to license AURORA-B only for conservative Appendix K LOCA analyses instead of best estimate plus uncertainty calculations, which simplified the review. Framatome has requested NRC approval of the AURORA-B LOCA methodology for operating BWRs, including power uprate and extended flow window conditions.

DISCUSSION

Codes and Methods

AURORA-B is based on four independent computer codes: MICROBURN-B2, a steady-state core simulator; RODEX4, a fuel thermal-mechanical code; S-RELAP5, a thermal-hydraulic system code; and MB2-K, a 3D transient neutron-kinetics code.

For LOCA applications, AURORA-B uses the point kinetics model built into S-RELAP5, instead of the MB2-K 3D neutronics. The staff finds this approach acceptable because neutron kinetics play a limited role in LOCA, which is mostly controlled by decay heat after scram.

These codes have individually received prior review and approval by the NRC for different applications. In particular, the RODEX4 fuel thermal-mechanical models address explicitly fuel thermal conductivity degradation and have been approved for use with all current Framatome fuels, including ATRIUM 11™ with chromium-doped fuel pellets.

The staff concentrated its review on the areas where modifications to the codes were made to accommodate the AURORA-B application to BWR LOCAs. These areas include: upper plenum mixing and spray condensation model; incorporation of BWR-specific fluid correlations; and code modifications to enforce Appendix K conservatisms. The staff performed TRACE confirmatory calculations to verify a number of AURORA-B assumptions and complete their review. The staff imposed additional conservatisms and limitations that are documented in the proprietary SE to ensure the code is applied appropriately within its validated ranges.

The staff concluded that, with the proposed limitations and conditions, the code modifications and methodology are acceptable. We concur with the staff evaluation of the AURORA-B LOCA methodology.

Qualification

Framatome has qualified the AURORA-B evaluation model using a large database of experimental data, including: five component effects tests, fifteen separate effects tests, and nine integral effects tests from three facilities with representative BWR conditions.

Framatome developed a phenomena identification and ranking table (PIRT) for BWR LOCA events. All the highly ranked phenomena were verified against the qualification database and AURORA-B provided conservative results.

Framatome did not address AURORA-B qualification for evaluation of long-term cooling. Instead, Framatome intends to use the existing generic evaluation of record by the reactor vendor. Consequently, the staff did not approve AURORA-B for long-term cooling evaluations.

Limitations and Conditions

The staff SE specifies 28 limitations and conditions to the applicability of the AURORA-B LOCA methodology. Most of them enforce limitations to use the methodology within the validated range of applicability using the agreed conservatisms. However, a number of limitations impose requirements for future licensees to justify the acceptability of a number of AURORA-B models on a plant-specific basis. The staff has informed us that they are exploring change-process guidelines that would apply to all vendors and would facilitate resolving these limitations on a generic basis. This is a good example of efforts by the staff to make the NRC more efficient while not losing focus on safety, and we encourage it. We look forward to interacting with the staff as these efforts mature.

SUMMARY

The staff's SE provides a comprehensive evaluation of this methodology and imposes limitations and conditions to ensure its appropriate application. It should be published.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

REFERENCES

1. AREVA, ANP-10332P, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios," Revision 0, February 2014 (ML14091A220).

2. U.S. Nuclear Regulatory Commission, "Audit Report for the May 16 to May 18, 2017, Audit in Support of the Review of ANP 10332P, Revision 0, 'AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios'," December 5, 2017 (ML17262B138).

3. Framatome, ANP-10300P-A, "AURORA-B: An Evaluation Mode for Boiling Water Reactors; Application to Transient and Accident Scenarios," Revision 1, January 2018 (ML18186A440 (Non-Public)/ ML18186A181 (Public)).

4. Framatome, ANP-10340P-A, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods," Revision 0, May 2018 (ML18171A107).

February 22, 2019

SUBJECT: SAFETY EVALUATION FOR ANP-10332P, REVISION 0, AURORA-B: AN EVALUATION MODEL FOR BOILING WATER REACTORS; APPLICATION TO LOSS-OF-COOLANT ACCIDENT SCENARIOS

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WASHINGTON, DC 20555 - 0001

February 21, 2019

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

SUBJECT: INTERIM LETTER: CHAPTERS 2 AND 17 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE NUSCALE SMALL MODULAR REACTOR

Dear Ms. Doane:

During the 660th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 2019, we met with representatives of NuScale Power, LLC (NuScale) and the NRC staff to review Chapter 2, "Site Characteristics and Site Parameters," and Chapter 17, "Quality Assurance and Reliability Assurance," 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 December 18, 2018. 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. We have not identified any major issues in Chapters 2 and 17 at this time. However, there are some items, as noted below, that need to be resolved.

2. The NuScale methodology for calculating accident offsite χ/Q values for the exclusion area boundary and low population zone coupled with the accident source term methodology for the NuScale design needs to be completed and reviewed by the staff.

3. The staff has requested an exemption from the Commission from requiring an inspection, test, analysis, and acceptance criterion, or ITAAC for the NuScale design reliability assurance program and this remains an open item.

4. The applicant's Open Design Items for structures, systems, and components covered by Chapter 17 requirements need to be identified for eventual closure.

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 its bases for acceptance of the DCA, the SER chapters also identify the information a combined license (COL) 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 aids in the resolution of concerns and facilitates timely completion of the design certification review. Accordingly, the staff has provided Chapters 2 and 17 of the SER with open items for our review. The staff's SER and our review of these chapters addressed DCA Chapter 2, Revision 1 and Chapter 17, Revision 1 and supplemental material, including NuScale responses to staff requests for additional information.

DISCUSSION

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

DCA Chapter 2 – Site Characteristics and Site Parameters

This chapter discusses the assumed site envelope for the NuScale small modular reactor (SMR) design and focuses on the geography and demography, nearby facilities, and postulated site parameters for the design, including meteorology, hydrology, geology, seismology, and geotechnical parameters. A COL applicant would have to demonstrate that their site falls within this assumed site envelope or demonstrate by other means that the proposed facility is acceptable at the proposed site.

The staff found that the NuScale approach to define the site envelope was acceptable with one open item to be resolved related to accidental radioactive releases. NuScale has revised its source term methodology, originally issued as TR-0915-17565. The staff is currently evaluating these revisions to the accident source term and the methodology for calculating the offsite χ/Q values used in determining the exclusion area boundary (EAB) and the low population zone (LPZ) in relation to the NuScale design or in a COL application referencing this design. The staff has checked the NuScale methodology and calculated χ/Q values, based on meteorological data collected at a number of nuclear power plant sites, assuming minimum EAB and LPZ outer boundary distances (400 feet). The results indicated that most of these sites were not bounded by the NuScale offsite χ/Q values. The staff anticipates that some COL applicants will require EAB and LPZ outer boundary distances larger than this minimum to be bounded by NuScale parameters.

DCA Chapter 17 – Quality Assurance and Reliability Assurance

This chapter describes the quality assurance (QA) program during the design phase, construction phase and operation phase. In addition, the chapter describes the reliability assurance program as it applies to safety-related and non-safety-related structures, systems, and components (SSCs) identified as being risk significant.

The reliability assurance program provides reasonable assurance that risk-significant SSCs identified in the final design are not degraded in operation and reliably function when challenged. It is a two-stage process. The first stage encompasses all activities that occur during the detailed design of the plant before initial fuel load; i.e., the design reliability assurance program (D-RAP). The second stage consists of the operational phase of the plant to ensure reliability of the SSCs during operations. This phase is left for the COL applicant to address.

Before certification, the D-RAP includes establishing the program, developing programmatic controls during design, and developing a D-RAP list of the SSCs using a defined methodology. This methodology (DCA Chapter 17, Figure 17.4.1) is based on a combination of probabilistic, deterministic, and other methods of analysis. The applicant used the probabilistic acceptance criteria defined in the approved NuScale licensing topical report, TR-0515-13952-A, on risk significance determination, to develop a candidate list of risk significant SSCs. Then the candidate list was reviewed by a NuScale panel of experts in risk analysis, safety, licensing, operation and maintenance. The panel evaluated and confirmed the D-RAP list. As the NuScale detailed plant design is completed, any new information about SSC reliability will be considered in the D-RAP evaluation process and the SSCs list would be updated as appropriate.

The staff reviewed the D-RAP list of risk significant SSCs and noted that the chemical and volume control system (CVCS) was not included. The CVCS provides an alternative means of borated reactor coolant makeup under accident conditions, which is diverse from the emergency core cooling system and provides defense-in-depth against core damage. The applicant noted that additional defense-in-depth is also provided by the containment flood and drain system in case the emergency core cooling system or CVCS are unavailable. The applicant noted that CVCS preoperational testing will be conducted and periodic operation of CVCS makeup pumps to adjust boron concentration and primary coolant inventory are part of normal operation, thus assuring an adequate means to determine component and system availability. The staff found the method of selecting the initial D-RAP SSCs to be appropriate. The final D-RAP SSCs including exclusion of the CVCS system will be confirmed after Chapter 19 is completed.

The staff noted that no ITAAC was identified to confirm completion of the D-RAP and this was listed as an open item in the safety evaluation. However, the staff noted that SECY-18-0093 was recently submitted to the Commission to request that this requirement be removed. This remains an open item to be resolved.

In addition, NuScale explicitly tracks "Open Design Items" (ODIs) as part of the engineering design process. ODIs are unverified engineering design assumptions that are part of engineering analyses used in the design process. ODIs may remain unverified until the affected SSCs are procured and required to be operable. NuScale has established a design control process to identify, track and close ODIs needed for the DCA. As part of the Chapter 17 review, the staff confirmed that this process meets the NRC quality assurance requirements (Criterion III to Appendix B of 10 CFR Part 50), although the 2017 staff inspection did identify instances in which the process could be improved during its implementation. NuScale indicated that identification and closure of ODIs, which they deemed necessary for the DCA, were closed. The staff has scheduled another inspection to confirm that the design control process is properly implemented. Those ODIs, which remain open after certification, will continue to be tracked by NuScale and closed by the cognizant design engineer. Those that can significantly affect safety analyses and probabilistic risk analysis results should be given priority and be made available for staff inspection.

SUMMARY

We have not identified any major issues in Chapters 2 and 17 at this time. However, there are some items that need to be resolved.

Sincerely,

/RA/

Peter C. Riccardella
Chairman

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5. Advisory Committee on Reactor Safeguards, "NuScale Power, LLC Licensing Topical Report, 'Risk Significance Determination'," May 18, 2016 (ML16130A373).

6. U.S. Nuclear Regulatory Commission, "Nuclear Regulatory Commission Inspection of the Quality Assurance Program Implementation Inspection of NuScale Power, LLC Report No. 05200048/2017-201," July 24, 2017 (ML17201J382).

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8. NuScale Power, TR-0515-13952-A, "Risk Significance Determination," Revision 0, July 2015 (ML15211A470).

February 21, 2019

SUBJECT: INTERIM LETTER: CHAPTERS 2 AND 17 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE NUSCALE SMALL MODULAR REACTOR

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DATE	2/21/2019	2/21/2019	2/21/2019	2/21/2019	2/21/2019

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

January 9, 2019

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

SUBJECT: EARLY SITE PERMIT – CLINCH RIVER NUCLEAR SITE

Dear Chairman Svinicki:

During the 659th meeting of the Advisory Committee on Reactor Safeguards (ACRS), December 6-7, 2018, we completed our review of the early site permit application submitted by the Tennessee Valley Authority (TVA) for two or more small modular reactors (SMRs) at its Clinch River Nuclear (CRN) Site, and the NRC staff's safety evaluation report. Our Regulatory Policies and Practices Subcommittee received an informational briefing on this topic on November 15, 2017, and also reviewed this matter at its meetings on May 15, August 22, October 17, and November 14, 2018. During our reviews, we had the benefit of discussions with the staff and representatives of TVA. We also had the benefit of the referenced documents. Our reviews of the application and the safety evaluation report were conducted to fulfill the requirements of 10 CFR 52.23, which states that the ACRS shall report on those portions of an early site permit application that concern safety.

CONCLUSION AND RECOMMENDATIONS

1. Small modular reactors with design characteristics within the plant parameter envelope used by TVA in developing its Clinch River Nuclear Site early site permit application can be constructed and operated without undue risk to the health and safety of the public.

2. The staff's safety evaluation report of the TVA early site permit application should be issued. The staff accepted TVA's plume exposure pathway emergency planning zone sizing methodology; two *major features* emergency plans (one plan for a site boundary plume exposure pathway emergency planning zone and a second plan for an approximate 2-mile radius plume exposure pathway emergency planning zone); and associated exemption requests. The safety evaluation report also identified a number of items that are treated either as permit conditions or as action items that must be addressed at the operating license stage.

3. The early site permit for the Clinch River Nuclear Site should be issued.

BACKGROUND

An early site permit is the Commission's approval of the safety and environmental suitability for a proposed site to support future construction and operation of one or more nuclear power plants. TVA's submittal addresses site suitability issues, environmental protection issues, and plans for coping with emergencies, independent of the review of a specific nuclear power plant design. Before a plant can be constructed, either under a combined license or a construction permit, a specific reactor technology for the site must be reviewed and approved by the NRC.

TVA filed an early site permit application for its CRN Site in May 2016 and the NRC accepted and docketed the application in December 2016. The TVA application was based on a plant parameter envelope (PPE) approach as a surrogate for a specific plant design. Using inputs from four prospective vendors (NuScale, Holtec, BWX Technologies, and Westinghouse) of light-water reactor-derivative SMR designs, TVA determined bounding values for construction and operation of two or more SMRs at the CRN Site with a total nuclear generating capacity up to 2420 MWt and 800 MWe (up to 800 MWt for a single unit or module). This approach allows TVA flexibility, while also potentially reducing licensing risk.

DISCUSSION

Proposed Site, Population, and Hazards Analyses

The proposed CRN Site encompasses 935 acres of land, bordered by the Department of Energy's Oak Ridge Reservation to the north and east, and by the Clinch River Arm of the Watts Bar Reservoir to the east, south, and west. Located within the City of Oak Ridge, Roane County, Tennessee, it is 6.8 miles east of Kingston, 9.2 miles east-southeast of Harriman, 8.8 miles northwest of Lenoir City, and 25.6 miles west-southwest of Knoxville, Tennessee. The land is owned by the U.S. Government and is managed by TVA as an agent of the federal government.

The exclusion area boundary is delineated by the boundaries of the CRN Property bordered by the Oak Ridge Reservation and the Clinch River. There are no residences, commercial activities, or traversing public roads and active railways within the exclusion area boundary. The low population zone is a one-mile radius from the center point of the site. Population density predictions for a 50-mile radius around the site are estimated at start of construction (2021), commencement of operations (2027), and through end of plant life (2067) to be well below siting guidelines (i.e., less than 500 people per square mile). The staff found the site information provided to be acceptable and meets the requirements of 10 CFR 100.20.

In general, potential hazards and accidents from nearby industrial, transportation, military, and aircraft operations were analyzed and were demonstrated to be well below frequency cut-offs and/or accidental dose guidelines. The staff in its evaluation of hazards set two permit conditions: one regarding main control room habitability for nearby transport of anhydrous ammonia and chlorine; and a second for the possible construction of a commercial airport in the nearby vicinity (about 6 miles from the site).

Site Characteristics

The CRN Site is well characterized in terms of geology, seismology, meteorology, and hydrology, and benefits from past site characterization (e.g., field meteorology measurements, borings, and excavation work) performed when the site was the location of the proposed, later

cancelled, Clinch River Breeder Reactor Project. The staff conducted site visits and audits, performed independent confirmatory calculations, and conducted thorough evaluations and reviews of each of these areas in the application. The staff concluded that the CRN Site characteristics meet the requirements of 10 CFR Part 100, "Reactor Site Criteria" and 10 CFR Part 20, "Standards for Protection Against Radiation." Subject to the safety evaluation report action items and permit conditions, there is reasonable assurance that approved reactor designs falling within the PPE design parameters for the CRN Site characteristics can be operated without undue risk to public health and safety.

Potential Radionuclide Releases

Radioactive Waste Management

TVA developed conservative PPE parameters for normal liquid and gaseous effluent release source terms for use in calculating offsite doses and used the LADTAP-II and GASPAR-II codes, respectively, to conduct exposure pathway dose analyses using site-specific hydrology and meteorology. The staff found that these analyses meet the design objectives of 10 CFR Part 50, Appendix I, environmental standards of 40 CFR Part 190, and dose limits of 10 CFR 20.1301. They concluded that reactor designs falling within the envelope of the PPE normal effluent release source terms and associated offsite doses are without undue risk to public health and safety. The staff issued an action item to verify that calculated doses to the public from normal effluent releases for the chosen reactor design are bounded by the doses evaluated in the early site permit. We concur with the staff's conclusions.

Accident Analyses

To evaluate offsite post-accident doses TVA selected the vendor-supplied design basis accident analyses with the highest post-accident doses for the site-specific dose analysis, and based the PPE source term on light-water reactor fuel representative of the SMR designs under consideration, assuming a single unit or module up to 800 MWt. Using site-specific short-term atmospheric dispersion factors (χ/Q methodology), TVA scaled the vendor-supplied doses and dispersion factors to obtain doses at the exclusion area boundary and low population zone boundaries. TVA was able to demonstrate that the surrogate plant would meet the requirements of 10 CFR 50.34(a)(1) and 52.17(a)(1): an individual at any point on the exclusion area boundary for any 2-hour period following the onset of fission product release would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE); and an individual located at any point on the outer boundary of the low population zone exposed to the radioactive cloud from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.

Consequences for bounding large-break loss-of-coolant accidents in SMRs are expected to be less than for large light-water reactors. TVA performed a comparison to similar analyses for the AP1000 plant (Vogtle 3 & 4 early site permit application) by scaling its thermal power by 0.235 (800MWt/3400 MWt). The scaled AP1000 dose result was 25% greater than the PPE surrogate for the worst 2-hour period, and roughly equivalent for a 30-day period, providing confidence in its analyses. The staff review found the analytical results adequate and acceptable in meeting the requirements of 10 CFR 50.34(a)(1) and 52.17(a)(1), and the PPE source term used not unreasonable in comparison to the AP1000 design. We concur with the staff's accident analysis assessment.

Emergency Preparedness

TVA proposed a risk-informed, dose-based, consequence-oriented methodology to determine the plume exposure pathway (PEP) emergency planning zone (EPZ). This would be consistent with the dose-savings approach developed in NUREG-0396 and used to meet the dose criteria of the Environmental Protection Agency (EPA) early-phase protective action guides (PAGs), (i.e., protection from doses above the 1 rem TEDE limit). The dose savings criteria of NUREG-0396 for determining the PEP EPZ are: 1) the EPZ should encompass those areas in which projected dose from design basis accidents could exceed the PAG; 2) the EPZ should encompass those areas in which the consequences of less severe core melt accidents could exceed the PAG; and 3) the EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in event of more severe core melt accidents (i.e., the conditional probability of exceeding 200 rem whole body dose outside the PEP EPZ is less than 1×10^{-3}).

For the first two criteria, an applicant would analyze design basis accidents and appropriate accident scenarios with a mean core damage frequency greater than 1×10^{-6} per reactor-year, determine source terms, calculate dose consequences, and compare results to the EPA early-phase PAG. For substantial reduction in early health effects, an applicant would use a core damage frequency of greater than 1×10^{-7} per reactor-year to select severe accident scenarios, then repeat the above process to calculate a distance at which the conditional probability to exceed 200 rem exceeds 1×10^{-3}.

Based on the above approach, and taking into consideration design information from the four SMRs, TVA developed two *major features*[1] emergency plans: one with the site boundary as the EPZ and a second with an approximate 2-mile radius EPZ. An evacuation time estimate study was also conducted for the 2-mile radius EPZ. The evacuation time estimate did not identify any physical characteristics unique to the site that would pose a significant impediment to development of future emergency plans.

At least one SMR design is expected to meet the dose criteria for the site boundary EPZ; all four are expected to meet the dose criteria for a 2-mile EPZ. TVA also developed a bounding, non-design-specific, composite, accident release source term for the PPE with a 25% added margin. Analyses demonstrate that the PEP EPZ criteria are met. The isotopic total release activity over 96 hours resulted in a TEDE of about 0.9 rem at the site boundary. Although we concur that the 96-hour time period was correctly implemented with the example calculations, it is important to select the most severe 96-hour period for the specific design.

TVA is seeking exemption requests to deviate from the 10-mile PEP EPZ [10 CFR 50.33(g) and 50.47(c)(2)], and from certain emergency planning requirements. To support their exemptions request, TVA cited anticipated enhanced safety features of the SMR designs considered: smaller radionuclide inventory and source terms, reduced likelihood of accidents, slower accident progression rates, and features to minimize or mitigate accident consequences.

TVA would then present a complete and integrated emergency plan with the combined license or construction permit application, based on the selected SMR technology and estimated dose consequences, resulting in either an EPZ at the site boundary, the approximate 2-mile radius, or an appropriately scaled EPZ. The ingestion pathway EPZ for the CRN Site would also be described in the application.

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[1] 10 CFR 50.47(a)(1)(iv)

The staff concluded that: TVA's PEP EPZ sizing methodology is acceptable because it is consistent with analyses that form the technical basis of the current 10-mile PEP EPZ and maintains the same level of protection (i.e., dose savings); the two *major features* emergency plans are acceptable; and the exemption requests are acceptable and will not present an undue risk to public health and safety. We concur with these staff conclusions.

SUMMARY

The TVA early site permit application and the staff's review demonstrated suitability of the CRN Site considering topics including surrounding population, external hazards, site physical characteristics, potential radionuclide releases, and emergency preparedness. This application is unique in its approach to emergency planning in that it proposes a risk-informed, dose-based, consequence-oriented methodology to determine the appropriate PEP EPZ. We note that this is in parallel to proposed rulemaking on emergency preparedness for small modular reactors and other new technologies, which we agreed with in our recent October 19, 2018 letter on this subject.

The TVA early site permit application benefits from the proposed use of advanced light-water reactor-derivative SMR designs that are expected to exhibit both lower accident frequencies and consequences than the current fleet of large light-water reactors; the large body of knowledge associated with light-water reactor technology, particularly regarding source terms; and extensive light-water reactor operating and licensing experience. TVA's approach to emergency planning in providing dose savings is consistent with that used in developing NUREG-0396 and the staff's proposed current rulemaking on the matter. The early site permit for the Clinch River Nuclear Site should be issued.

Sincerely,

/RA/

Michael L. Corradini
Chairman

REFERENCES

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January 9, 2019

SUBJECT: EARLY SITE PERMIT – CLINCH RIVER NUCLEAR SITE

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

December 19, 2018

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE SEABROOK STATION, UNIT 1

Dear Chairman Svinicki:

During the 659th meeting of the Advisory Committee on Reactor Safeguards (ACRS), December 6-7, 2018, we completed our review of the license renewal application (LRA) for the Seabrook Station, Unit 1 (Seabrook), and the final safety evaluation report prepared by the NRC staff. Our Plant License Renewal Subcommittee reviewed this matter during meetings on July 10, 2012, October 31, 2018, and November 15, 2018. During these reviews, we had the benefit of discussions with representatives of the staff and NextEra Energy Seabrook, LLC (NextEra). We also had the benefit of the referenced documents. This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

1. The programs established and committed to by NextEra to manage age-related degradation provide reasonable assurance that Seabrook can be operated in accordance with its licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. NextEra's application for renewal of the operating license for Seabrook should be approved.

BACKGROUND

Seabrook is located in the town of Seabrook, Rockingham County, New Hampshire, on the western shore of Hampton Harbor, two miles west of the Atlantic Ocean. Seabrook is approximately two miles north of the Massachusetts state line and approximately 15 miles south of the Maine state line.

Seabrook is a single unit Westinghouse 4-loop pressurized water reactor with a General Electric turbine generator. Seabrook's licensed core power level is 3468 megawatts thermal with a net power output of approximately 1245 megawatts electric. A zero power license was granted to

the facility in October 1986 and a full power operating license was granted on March 15, 1990. Seabrook sought and received a modification to the expiration of the facility operating license to recapture the time licensed at zero percent power.

Originally two identical units were to be built on the site. Construction of Seabrook Station Unit 2 was effectively terminated in 1984 when it was approximately 25 percent complete and the construction permit subsequently expired in October 1988.

In this application NextEra requests renewal of the operating license for Seabrook (Facility Operating License NPF-86) for a period of 20 years beyond expiration of its current license. This would extend the operating license from midnight, March 15, 2030, to midnight, March 15, 2050.

DISCUSSION

In preparation for life extension, NextEra completed improvements, upgrades, replacements, and modifications to numerous systems and components. These include vital batteries, vital inverters, generator step-up transformers, service water piping, incore detectors, process control single point vulnerability circuit cards, solid state protection system circuit cards, rod control motor/generator sets, and shutdown reactor coolant pump seals. The Mechanical Stress Improvement Process® was completed for all reactor vessel nozzles.

In its final safety evaluation report, the staff documented its review of the LRA and other information submitted by NextEra and obtained through staff audits and inspections at the plant site. The staff reviewed the completeness of the identification of structures, systems, and components that are within the scope of license renewal. The staff also reviewed the integrated plant assessment process; the identification of plausible aging mechanisms associated with passive, long-lived components; the adequacy of the Aging Management Programs (AMPs); and the identification and assessment of Time-Limited Aging Analyses (TLAAs).

The LRA identified the structures, systems, and components that fall within the scope of license renewal. While the original application was prepared in accordance with the Generic Aging Lessons Learned (GALL) Report, Revision 1, it was updated and it now demonstrates consistency with the Generic Aging Lessons Learned (GALL) Report, Revision 2, and justifies deviations to the specified approaches in that report. The Seabrook AMPs are implemented in accordance with appropriate elements of the requirements of 10 CFR Part 50, Appendix B, specifically corrective actions, confirmation process, and administrative controls. The Seabrook Quality Assurance Program applies to safety-related structures and components. In its review, the staff concluded that NextEra's quality assurance program application was adequate to ensure that LRA associated activities were performed in accordance with NextEra's license renewal program requirements.

NextEra will implement 44 AMPs for license renewal, comprised of 29 existing programs and 15 new programs. Of the 15 new programs, seven are consistent with the GALL Report, one is consistent with enhancement, two are consistent with allowed exceptions, one is consistent with enhancements and allowed exceptions, and four (Buried Piping and Tanks Inspection, 345 KV SF_6 Bus, Building Deformation Monitoring, Alkali-Silica Reaction (ASR) Monitoring) are plant specific. Of the 29 existing programs, eight are consistent with the GALL Report, 12 are consistent with enhancements, two are consistent with allowed exceptions, five are consistent with enhancements and allowed exceptions, and two (Nickel Alloy Nozzles and Penetrations,

Boral Monitoring) are plant-specific. The LRA includes five programs with allowed exceptions to the GALL Report. The programs with exceptions and enhancements are acceptable.

Two of the new plant specific AMPs focus on concrete degradation caused by ASR. The Building Deformation Monitoring Program and the Alkali-Silica Reaction (ASR) Monitoring Program address NextEra's approach to assess, monitor, and manage ASR. In a separate letter report dated December 14, 2018, we provide our specific findings and conclusions related to this issue.

The staff conducted license renewal audits and performed a license renewal inspection at Seabrook. The audits verified the appropriateness of the scoping and screening methodology for AMPs, the appropriateness of the aging management review, and the acceptability of the TLAAs. The staff audit report demonstrated the validity of its conclusion that the Seabrook Aging Management Program is mature. The NextEra organization is benefiting from their fleet approach that integrates lessons-learned from each facility. The license renewal inspection verified that the license renewal requirements are implemented appropriately. The audits and inspections were comprehensive and the corresponding reports are thorough.

Based on these audits, inspections, and the staff reviews, the staff concluded that NextEra has demonstrated that the effects of aging at Seabrook will be adequately managed so that the intended safety functions will be maintained consistent with its licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff's review of the LRA identified no open or confirmatory items. We agree with the staff conclusion that there are no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) that preclude renewal of the operating license for Seabrook.

SUMMARY

The programs established and committed to by NextEra to manage age-related degradation provide reasonable assurance that Seabrook can be operated in accordance with its licensing basis for the period of extended operation without undue risk to the health and safety of the public. The NextEra application for renewal of the operating license for Seabrook should be approved.

Member Riccardella did not participate in deliberations on this topic.

Sincerely,

/RA/

Michael Corradini
Chairman

REFERENCES

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10. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.188, "Standard Format and Content for Application to Renew Nuclear Power Plant Operating Licenses," Revision 1, September 2005 (ML082950585).

December 19, 2018

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE SEABROOK STATION, UNIT 1

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

December 14, 2018

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

SUBJECT: SEABROOK STATION UNIT 1 LICENSE RENEWAL APPLICATION: REVIEW OF LICENSEE PROGRAM ADDRESSING ALKALI-SILICA REACTION

Dear Chairman Svinicki,

During the 659th meeting of the Advisory Committee on Reactor Safeguards (ACRS), December 6-7, 2018, we completed our review of the license renewal application (LRA) for the Seabrook Station, Unit 1 (Seabrook) submitted by NextEra Energy Seabrook, LLC (NextEra). Our review considered NextEra's actions to address a concrete degradation mechanism observed in plant structures, known as alkali-silica reaction (ASR). Degradation typical of ASR was first detected at the plant in 2009, and confirmed by concrete borings withdrawn from Seabrook structures in 2010. Since that time, NextEra has undertaken substantial and thorough actions to identify, understand, and address this condition. In August 2016, NextEra submitted License Amendment Request (LAR) 16-03 to revise the Seabrook current licensing basis to adopt a methodology for the analysis of Seismic Category I structures with concrete affected by ASR.

To conduct a focused review of past, current, and future actions to address ASR at Seabrook, our Plant License Renewal Subcommittee met with the NRC staff, NextEra and their consultants on October 31, 2018, separately from our general Seabrook license renewal subcommittee meeting. We also had the benefit of the referenced documents. This letter summarizes that review.

CONCLUSIONS

1. NextEra License Amendment Request 16-03 establishes a robust analytical methodology, supported by a comprehensive large scale test program, for the treatment and monitoring of alkali-silica reaction-affected Seismic Category I structures at Seabrook.

2. The NextEra license renewal application includes two new Aging Management Programs to monitor alkali-silica reaction and building deformation. These incorporate the test program results and license amendment request methodology and assure that the effects of alkali-silica reaction will be effectively tracked and evaluated through the end of the license renewal application period of extended operation.

3. The staff safety evaluations of the license amendment request and alkali-silica reaction-related Aging Management Programs in the license renewal application provide thorough assessments and findings. We agree with the staff's conclusion that NextEra's programs are acceptable.

BACKGROUND

Alkali-silica reaction occurs in concrete, in the presence of moisture, when reactive silica in the concrete aggregate reacts with alkali ions in the pore solution. The reaction produces an alkali-silica gel that expands in volume as it absorbs moisture, resulting in cracking of the concrete and potentially reducing the capacity of concrete structures. The ASR discovered at Seabrook is a slowly developing phenomenon that was initially manifested as micro-cracking and staining of concrete structures. In this instance, its presence is the result of moisture combined with chemically reactive aggregate used in plant construction that was mined from a quarry approximately 150 miles from the site. Based on data and testing performed on the Seabrook concrete, it is reasonable to assume that the ASR phenomenon has been occurring since early operation of the plant (although virtually undetectable in its early stages) and will continue to occur through the balance of plant life.

Because ASR affects concrete properties and imposes structural loadings that were not originally addressed in Seabrook's operating license basis assessments, NextEra submitted LAR 16-03 to revise the Seabrook Updated Final Safety Analysis Report to include methods for analyzing Seismic Category I concrete structures affected by ASR. The LAR is based on testing and analyses that established appropriate concrete properties and analytical methods to demonstrate the acceptability of structures considering the effects of ASR. The LAR methodology has been used to analyze all Seismic Category I structures at Seabrook in their current, ASR-degraded condition, as well as to develop plant specific Aging Management Programs (AMPs) in the LRA to demonstrate that the structures affected by ASR will be acceptable for the proposed period of extended operation (PEO). The staff has completed their reviews of the LAR and ASR-related AMPs in the LRA, and documented their findings in its safety evaluations. We examine each of these topics further in the subsequent sections.

DISCUSSION

Large Scale Testing Program

At the time of initial discovery of ASR at Seabrook, limited data were available addressing ASR in highly constrained (reinforced) concrete such as that used in Seabrook Seismic Category I structures. These structures employ tightly spaced, two-dimensional reinforcing grids that restrict the ability of the concrete to expand in the plane of the grids. NextEra thus conducted a multi-year, large scale testing program (LSTP) at the University of Texas Ferguson Structural Engineering Laboratory. This program determined that, although ASR causes significant degradation of concrete strength properties as measured by standard core samples removed from ASR-degraded structures, the highly reinforced structures themselves did not experience an associated reduction in structural capacity. In fact, the presence of ASR actually increased the load-carrying capacity of such structures. This was a highly repeatable phenomenon, observed through numerous tests on large-scale specimens, fabricated with concrete intentionally subjected to accelerated ASR, well beyond the levels observed to date at Seabrook. Testing included shear capacity tests, reinforcement anchorage and beam flexure tests, as well as concrete anchor pullout tests in ASR-affected concrete.

The LSTP test samples were highly representative of the ASR-affected structures at Seabrook. They incorporated prototypical characteristics for structural dimensions, reinforcing bar sizes and spacing, concrete aggregate, unreinforced concrete cover, and concrete compressive strength prior to ASR. Sodium hydroxide was added to the test sample concrete mixtures to accelerate ASR, which, in conjunction with environmentally controlled aging, enabled the samples to exceed projected plant-level ASR expansion (including the LRA PEO) in a comparatively short timeframe.

ASR Analysis Methodology

In August 2016, NextEra submitted LAR 16-03 to revise the Seabrook current licensing basis to adopt a methodology for the analysis of Seismic Category I structures with concrete affected by ASR. The LSTP results were used to develop a set of analysis guidelines to address ASR in a manner consistent with the original industry standard codes used in design and construction of these structures at Seabrook (ASME Section III and ACI 318-71).

Notably, the revised analysis guidelines specify that original (non-ASR-degraded) concrete material strength properties be used in the analyses. Although the LSTP results described above demonstrated that ASR enhanced the structural capacity measurements, ASR also increases demand on the structures, in the form of increased compressive stresses in the concrete and increased tensile loads in the steel reinforcing members, due to volumetric expansion of the concrete. Methods for incorporating these new loadings, using appropriate load factors, are addressed in the LAR.

NextEra also commissioned detailed reanalysis of all ASR-affected Seismic Category I structures, as well as the intake and discharge structures, adding ASR loading to existing design basis loads, in accordance with the LAR methodology. This methodology will also be used to establish building-specific, acceptable margins and building deformations as ASR progresses during the PEO.

Augmented Structural Monitoring Program

The LSTP also included tests to define instrumentation and measurement techniques that can be used to monitor current and future development of ASR. The testing determined that ASR expansion in the plane of the reinforcing grids (in-plane expansion) saturates at a relatively low strain level, after which the expansion is constrained by the reinforcement. Subsequent expansion then occurs transverse to the reinforcing grids (out-of-plane expansion). The testing identified different measurement techniques for monitoring these two modes of ASR expansion. The test program and analyses determined that in-plane expansion can be effectively monitored using a method of crack width monitoring within defined grid patterns on the concrete surfaces (combined cracking index or CCI). After the in-plane expansion saturates, subsequent out-of-plane expansion will be characterized using a specially developed snap ring borehole extensometer. The LSTP defined the accuracy and limitations of these two measurement techniques, which have been incorporated into structural monitoring AMPs for ASR at Seabrook. An ASR monitoring AMP defines measurement locations, measurement time intervals, and ASR expansion limits beyond which corrective action is required.

A second new AMP establishes a building deformation monitoring program to monitor gross building deformation as the result of ASR and compare it to acceptable limits. Predicted building deformations from these analyses are compared to periodic measurements to evaluate their accuracy and to ensure that building deformations do not exceed acceptable margins. The

monitoring program also ensures that required building-to-building seismic gaps and seal dimensions are maintained, and that relative building deformations do not damage interconnecting piping and other equipment.

Independent ASR Research

As previously noted, at the time of initial ASR observation at Seabrook in 2009, there were limited data available on the effects of ASR on highly constrained structures. However, in the interim, a large body of ASR research similar to the LSTP is ongoing, including domestic research sponsored by the NRC's Office of Nuclear Regulatory Research at the National Institute of Standards, as well as large Canadian and European programs. These programs have produced similar results to the LSTP, observing increased structural capacity in highly-constrained, ASR-affected structures. Also noteworthy is the fact that the National Institute of Standards program (and others) chose a similar approach of fabricating prototypical, structural-sized test samples, with concrete produced to artificially accelerate ASR.

Staff Reviews

The staff has dedicated continuous regulatory oversight to the licensee programs addressing ASR issues at Seabrook, from initial and continued operability determinations with the ASR condition to review and approval of the AMPs for structural and building deformation monitoring. Since the initial observation of ASR at Seabrook, the staff has maintained a strong, augmented program of inspection and audits to identify plant condition changes due to ASR, as well as structures and equipment functions that may be affected by such conditions. Several important ASR-related conditions were identified by staff auditors or regional inspectors, either as a direct result of walkdowns or by their detailed reviews of corrective actions, licensee condition reports, and program bases documents. These included:

- Groundwater infiltration as a potential cause of accelerated degradation of plant concrete and steel support structures
- Water leakage from the spent fuel pool fuel transfer canal
- Structure or building deformation caused by widespread ASR expansion

The slow progression of the ASR conditions at Seabrook allowed NextEra to develop appropriate experimental and analytical approaches to derive better understanding and predictive capability of ASR impacts via their LSTP. The staff's review and audit process for this test program confirmed that the research, testing, analyses, and application of results met industry quality standards and NRC regulatory requirements. In parallel, the staff established frequent, periodic audits and team reviews of each of the ASR-related response programs at the site. The reviews demonstrated that the NextEra organization was fully prepared to implement and execute these programs.

The staff's review process for the NextEra LSTP, LAR, and ASR programs in the LRA has been deliberate and comprehensive, and it included effective use of requests for additional information to identify and resolve critical issues. This has resulted in robust analytical procedures and AMPs that are well documented in the final NextEra LAR and LRA submittals. The staff's LAR and LRA safety evaluations and their referenced reports document their audits and reviews and provide thorough assessments of the NextEra programs designed to assure proper identification, monitoring, and evaluation of ASR-related conditions at Seabrook. The

staff assessments conclude that these programs will effectively identify and characterize the ASR condition and are capable of evaluating its impact on the ability of the affected structures to accomplish their design basis functions.

SUMMARY

NextEra has undertaken comprehensive actions to characterize, evaluate, and apply test results into Seabrook-specific analysis and monitoring programs to understand current building structural capacity and to monitor and evaluate future building performance. The staff has conducted assessments of the testing program, the data from the testing, and the efficacy of licensee employment of these programs as bases for judging the acceptability of the affected structures for present and extended life through the PEO. We concur with the staff conclusion that, while some of the structures are degraded, they are fully capable of performing their credited function through the requested PEO under the committed enhanced monitoring and evaluations.

Sincerely,

/RA/

Michael Corradini,
Chairman

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December 14, 2018

SUBJECT: SEABROOK STATION UNIT 1 LICENSE RENEWAL APPLICATION: REVIEW OF LICENSEE PROGRAM ADDRESSING ALKALI-SILICA REACTION

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