July 16, 2024

The Honorable Christopher T. Hanson
Chair
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SUMMARY REPORT – 716th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JUNE 5-6, 2024

Dear Chair Hanson:

During its 716th meeting, June 5 through 6, 2024, which was conducted in person and virtually, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters. The ACRS completed the following correspondence:

LETTERS

Letter to Raymond V. Furstenau, Acting Executive Director for Operations (EDO), U.S. Nuclear Regulatory Commission (NRC) from Walter L. Kirchner, Chair, ACRS:

- Draft Safety Evaluation of TerraPower Topical Report, “Principal Design Criteria for the Natrium Advanced Reactor,” Revision 1, dated June 28, 2024, Agencywide Documents Access and Management System (ADAMS) Accession No. ML24170A853, and

- Draft Safety Evaluation of TerraPower’s Natrium Topical Report on Fuel and Control Assembly Qualification, dated June 27, 2024, ADAMS Accession No. ML24172A046.

MEMORANDUM

Memorandum to Raymond V. Furstenau, Acting EDO, NRC, from Scott W. Moore, Executive Director, ACRS:

- Documentation of Receipt of Applicable Official NRC Notices to the Advisory Committee on Reactor Safeguards for June 2024, dated June 13, 2024, ADAMS Accession No. ML24163A271.

HIGHLIGHTS OF KEY ISSUES

a. Draft Safety Evaluation of TerraPower Topical Report, “Principal Design Criteria for the Natrium Advanced Reactor,” Revision 1

The Committee heard from licensee representatives and NRC staff, and it issued a June 28, 2024, letter, with the following conclusions and recommendations:

1. The Principal Design Criteria (PDC) proposed by TerraPower adapt the advanced reactor design criteria provided in NRC guidance to the Natrium plant design. Additionally, TerraPower stated they continue to assess the potential for additional PDC as part of the licensing process.

2. As we stated in 2018, we support application of the functional containment concept to advanced reactors like Natrium and note that establishing the details of a functional containment can be complex. TerraPower has not yet fully developed the Natrium functional containment design, and the concept described by TerraPower results in a set of barriers that share many similarities with previous sodium-cooled fast reactor (SFR) designs. Therefore, the functional containment approach is expected to maintain significant defense-in-depth capability comparable to prior SFRs.

3. The Safety Evaluation Report (SER) should be issued.

b. Draft Safety Evaluation of TerraPower's Natrium Topical Report on Fuel and Control Assembly Qualification

The Committee heard from licensee representatives and NRC staff, and it issued a June 27, 2024, letter, with the following conclusions and recommendations:

1. The proposed operating envelope for Natrium fuel compared to the historical fast reactor metallic fuel performance database provides confidence that the fuel will perform with adequate margin under both normal operation and transient overpower conditions.

2. Quantitative fuel design limits that are tied to the regulatory acceptance criteria remain to be established as the design and safety analysis evolve.

3. The fuel surveillance program is important because it provides a way to safely manage the operation of Natrium fuel in light of any residual performance uncertainties once all the testing and code validation activities are complete.

4. The mechanical restraint system with the associated load pads is unique. The applicant has committed to monitoring during operation to validate the radiation effects in the modeling predictions of this system. Given this system's importance in reactivity control, a detailed plan for this activity is recommended for review and approval by the NRC prior to initial operation.

5. As recognized above, significant work remains to provide all of the information necessary to qualify the fuel assemblies, control assemblies and mechanical core restraint system in Natrium. These limitations are appropriately noted by the staff. The SER with its associated limitations and conditions should be issued.

c. Discussions at the Planning and Procedures Session

1. The Committee discussed the Full Committee (FC) and Subcommittee (SC) schedules through November 2024 as well as the planned agenda items for Full Committee meetings.

2. The ACRS Executive Director led a discussion of significant notices issued by the Agency since the last Full Committee meeting in May 2024. The Executive Director documented this activity in a memorandum dated June 13, 2024, ADAMS Accession No. ML24163A271.

3. The Executive Director noted that there were no regulatory guides (RG)/draft regulatory guides to review this month.

4. Vice Chair Halnon led a discussion of the planned trip to Region II sites for the week of July 22, 2024. This visit will include a visit to a Tennessee Valley Authority operating plant (Sequoyah Nuclear Plant, Units 1 and 2), the Technical Training Center, and Region II offices including a Plant Operations Subcommittee meeting while at Region II. Additional logistical information will be provided to members in the future.

5. Member-at-Large Petti led a discussion of the planned SC and FC meetings to support the ACRS review of the Kairos Hermes 2 construction permit application. The SC meeting that was scheduled for June 12 was no longer needed and the SC will meet on June 26, 2024, to perform preparatory work for the final Committee letter that is scheduled for the July 2024 FC meeting.

6. Vice Chair Halnon led a discussion about the Holtec Quality Assurance Program Description (QAPD) Topical Report (TR). The TR provides for mandatory programmatic requirements under the applicable eighteen criteria in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B and American Society of Mechanical Engineers (ASME) NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications." The TR is arranged in parallel with the criteria and laid out traditionally as expected. Holtec provides design, analysis, manufacturing, decommissioning, and site services to a variety of industry sectors using its Holtec Quality Assurance Manual (HQAM). This HQAM complies with 10 CFR Part 21; 10 CFR Part 50, Appendix B; 10 CFR Part 71, Subpart H; and 10 CFR Part 72, Subpart G in addition to ASME NQA-1 with stated exceptions. The TR provides the subset of HQAP requirements for design, fabrication, construction, and testing activities necessary for the Holtec small modular reactor (SMR). It complies with ASME NQA-1-2015, with only an administrative exception for lead auditor qualifications. There is no concern with the scope or exceptions, therefore Member Halnon did not recommend review by the ACRS. The Committee agreed with the recommendation.

7. Vice Chair Halnon led a discussion about the Committee's activities associated with the potential restart of the Palisades plant. He mentioned that there is an information session on this topic scheduled for the October 2024 FC meeting.

8. Member Ballinger led a discussion of the Electric Power Research Institute (EPRI), "BWRVIP-100, Revision 2: BWR Vessel and Internals Project: Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Internal Components," TR Review. This document presents an update of the available fracture toughness data

for irradiated stainless steel used for boiling water reactor (BWR) internals, primarily the core shroud. One of the staff recommendations during its review of BWRVIP-100, Revision 1 (BWRVIP-100, Revision 1-A), was that additional fracture toughness data should be obtained to strengthen the existing data base. This report documents acquisition of additional data and the incorporation of this data into the evaluation methodology and provides an updated fracture toughness correlation. Revised correlations have been developed for both weld and base metal.

Revision 1 has previously been reviewed, and approved, by the staff. Revision 2 provides additional data and a revised correlation. There is thus no new safety-related information. Member Ballinger recommended that the Committee not review the TR. The Committee agreed.

9. Chair Kirchner led a discussion of the General Electric (GE) Hitachi Nuclear Energy's BWRX-300 TR on Steel-plate Composite Containment Vessel (SCCV) and Reactor Building (RB) Structural Design.

   The BWRX-300 integrated RB consists of the RB structure enclosing the containment, the containment structure comprised of the SCCV, containment closure head and other ASME Class MC components, and the containment internal structures. The integrated RB is the only BWRX-300 Seismic Category I structure.

   The BWRX-300 integrated RB is constructed using steel-plate composite modules to maximize its safety performance during the operational and decommissioning life of the plant and to optimize the construction cost and schedule. The BWRX-300 integrated RB is deeply embedded so that the majority of the reactor pressure vessel, SCCV structure, and other important safety-related systems and components are located below grade to mitigate the effects of possible external events, including aircraft impact and adverse weather.

   Current design codes do not address the use of steel-plate composite *systems as a containment pressure boundary. Therefore, design rules for the SCCV are proposed in Section 6.0 that are based on the ASME BPVC, Section III,* Rules for Construction of Nuclear Facility Components, Division 2, Code for Concrete Containments, Subsection CC, Concrete Containments, Articles CC-1000 through CC-6000, for materials, design, fabrication, construction, examination and testing for the BWRX-300 SCCV, *including Division 2 Appendices to the extent they apply to a* steel-plate composite *containment without reinforcing steel or tendons*. Design rules for the RB and containment internal structures that are not part of the containment pressure boundary follow existing codes and standards for design of steel-plate composite structures with proposed modifications provided in Section 5.0 to cover design elements beyond the scope of current codes and standards.

   The steel-plate composite modules used in the construction of the BWRX-300 integrated RB consist predominantly of steel-plate composite modules with diaphragm plates (see Section 3.4 for details). The proposed design approaches for the RB, containment internal structures and SCCV using these steel-plate composite modules are supplemented by a test program that is being performed under the National Reactor Innovation Center (NRIC) Advanced Construction Technology (ACT) project in the United States. This program is known as the NRIC Demonstration Project and is described in Section 7.0.

Chair Kirchner proposed that ACRS review NEDC-33926P/NEDO-33926, Revision 1, “BWRX-300 Steel-Plate Composite Containment Vessel (SCCV) and Reactor Building (RB) Structural Design.” The Committee agreed. A SC meeting has been scheduled for July 9, 2024.

10. Member Ballinger led a discussion on the EPRI report on motor operated valve performance prediction methodology (PPM), Version 4.1. This document describes a computer program that estimates the performance of Motor Operated Valves. Member Ballinger recommended that the Committee not review this document. The Committee agreed.

11. Member Ballinger led a discussion on the review of EPRI MRP-227, “Materials Reliability Program: PWR Internals Inspection and Evaluation Guidelines,” Revision 2.

    MRP-227 provides detailed guidance for the inspection and evaluation of PWR internal structural components subject to long term aging. Revision 2 provides an extensive update for operating experience to address subsequent license renewal materials degradation issues.

    In addition to these updates, this revision adds three new appendices that provide guidance in implementation of alternate aging management approaches, other than inspection and evaluation.

    Appendix C: Options for Alternate Aging Management Approaches for Westinghouse and Combustion Engineering (CE) Designs

    Appendix D: Guidance for Flexible Power Operation of Westinghouse and CE Designs

    Appendix E: Incorporation of Interim Guidance from MRP-191, Revision 2, “Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and CE PWR Designs (Interim Guidance from MRP 2018-022)

    Appendix C describes a proactive replacement or modification strategy as an alternative for an increasing number of inspections and/or reducing the risk of unexpected degradation. Alternate strategies include extensive repairs or modifications, component replacement or remote condition monitoring. Appendix C provided detailed guidelines for inclusion and effect of these alternatives in plant Aging Management Programs.

    Pressurized water reactors were originally designed for base-load operation. However, the incorporation of wind and solar sources has forced plants to load follow. Appendix D addresses the effect of non-baseload operation on the reactor vessel internals aging management program for Westinghouse and Combustion Engineering plants. Babcock and Wilcox (B&W) designed plants are outside MRP-227 guidance and require plant specific guidance.

    Appendix E provides updated guidance that addresses the extension of operation from 60 years to subsequent periods of operation. Key differences from previous revisions include a separation of degradation consequences into safety and economic categories.

Electric Power Research Institute MRP-227, Revision 2 represents a significant update from previous revisions. The period of extended operation (including load-following) has had a significant impact on age-related degradation management. For this reason, Member Ballinger recommended the Committee review the document. The Committee agreed.

12. Member Ballinger led a discussion about the details concerning the site visit by several members to the GE-Hitachi nuclear fuel fabrication facility scheduled for September 17, 2024. The ACRS staff will provide additional information on logistics to support the trip.

13. The Chair led a discussion about the logistics regarding the Commission meeting with the ACRS that is scheduled for June 7, 2024, at 10 a.m.

14. There was a closed portion of the meeting held to discuss personnel and administrative issues.

15. The following topics are on the agenda of the 717th ACRS FC meeting scheduled for July 10 through 12, 2024:

- Final letter report for the Kairos Hermes 2 construction permit application, and

- Research topic on risk assessment and human factors for non-light water reactors.

Sincerely,

Walter L. Kirchner Signed by Kirchner, Walter on 07/16/24

Walter L. Kirchner
Chair

July 16, 2024

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