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

January 13, 2025

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

SUBJECT: SUMMARY REPORT – 721st MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, DECEMBER 4, 2024

Dear Chair Hanson:

During its 721st meeting on December 4, 2024, which was conducted in person and virtually, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters. The ACRS completed the following correspondence:

MEMORANDA

Memoranda to Dr. Mirela Gavrilas, Executive Director for Operations, U.S. Nuclear Regulatory Commission (NRC), from Scott W. Moore, Executive Director, ACRS:

- Documentation of Receipt of Applicable Official NRC Notices to the Advisory Committee on Reactor Safeguards for December 2024, dated December 12, 2024, Agencywide Documents Access and Management System (ADAMS) Accession No. ML24345A163, and
- Regulatory Guides (RGs), dated December 12, 2024, ADAMS Accession No. ML24345A081.

HIGHLIGHTS OF KEY ISSUES

a. Material Reliability Program: Pressurized Water Reactor (PWR) Internal Inspection and Evaluation Guidelines (MRP-227, Revision 2)

Member Ballinger led a discussion of the Fuels, Materials, and Structures Subcommittee (SC) meeting that was held on this topic on November 21, 2024, (the transcript and slides may be found at ADAMS Accession No. ML24337A166).

The SC recommended no letter report and offered the following for inclusion in the summary report to document its review.

This MRP-227 revision provides detailed, updated, guidance for the inspection and evaluation of PWR internal structural components, subject to NRC approved long-term aging management programs for Subsequent License Renewal applications. Revision 2 provides an extensive update that incorporates lessons from 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 the 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," 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 incorporation of these alternatives into plant Aging Management Programs.

Appendix D, "Guidance for Flexible Power Operation of Westinghouse and CE Designs," notes that flexible power operation is, in effect, load following in discrete steps that is not the same as continuous small power changes in response to requests from the grid operator. The current fleet of PWRs were primarily designed for base-load operation. However, the incorporation of wind and solar sources has forced plants to adapt. 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 designed plants are outside MRP-227 guidance and require plant specific guidance.

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)'," provides updated guidance that addresses the extension of operation from 60 years to subsequent periods of operation.

The MRP-227, Revision 2 represents a significant update from previous revisions and reflects effective assimilation of operating experience into internals degradation management. The safety evaluation (SE) has been approved by the staff with no limitations and conditions. The SC concurs with the SE.

Operating plants that have chosen to adopt the recommendations in MRP-227A, Revision 2 are bound by their licensing basis. The SC noted that plants that have ceased operations and entered decommissioning, but wish to restart, are not necessarily covered with the current version of MRP-227-soon to be approved as MRP-227A, Revision 2. For plants that wish to restart, extenuating conditions such as internals layup, may not be adequately addressed by MRP-227 (version of record) and would need to further justification for the use of MRP-227 (version of record) for periods of extended operations after License Renewal or Subsequent License Renewal approval. An update of MRP-227, Revision 2 in the form of industry interim guidance could be developed for these plants or it can be addressed plant-specifically by individual licensees at the time of their license renewal application.

After some discussion, the Committee agreed with the SC recommendations including inclusion of the documentation in the summary report.

b. Discussions During the Planning and Procedures Session

1. The Committee discussed the Full Committee (FC) and SC schedules through May 2025 as well as the planned agenda items for FC meetings.

2. The ACRS Executive Director led a discussion of significant notices issued by the Agency since the last Full Committee meeting in November 2024. The Executive Director documented this activity in a memorandum dated December 12, 2024, ADAMS Accession No. ML24345A163.

3. The Executive Director also led a discussion of two draft RGs regarding possible review by the Committee. The Executive Director documented this activity in a memorandum dated December 12, 2024, ADAMS Accession No. ML24345A081.

4. Member Roberts led a discussion of the reconciliation of Revision 9 of Standard Review Plan Branch Technical Position (BTP) 7-19, "Guidance for Evaluation of Defense in Depth and Diversity to Address Common-Cause Failure Due to Latent Defects in Digital Safety Systems." This was a continuation of a discussion from the November 2024 FC meeting. The Committee agreed that the following would suffice as documentation on this issue.

   During the 721st meeting of the ACRS, December 4, 2024, the Committee discussed the staff response to the ACRS letter dated March 22, 2024 (ADAMS Accession No. ML24075A286). The Committee considers that the response to Recommendation No. 3 regarding the desirability of additional guidance for determining adequacy of defense in depth for digital instrumentation and control (I&C) in light water reactors partially addresses the recommendation. While the response states that the 1994-era "echelons of defense" guidance previously cited in BTP 7-19 is obsolete, it does not discuss what principles are currently in use to replace the obsolete guidance or whether replacement guidance is warranted.[1]

   Further review of principles or guidance for digital I&C "echelons of defense" would be most appropriately conducted in the context of a proposed design. NRC staff is currently reviewing a digital I&C upgrade for the Limerick nuclear generating station, in which the applicant proposes integrating numerous existing analog systems into one digital system. The bases used by NRC staff to conclude that this upgrade adequately supports defense in depth would provide a good case study to inform a discussion of modern application of the digital I&C "echelons of defense." The Committee recommends that the staff work with the ACRS staff to determine the appropriate timing for a meeting of the digital I&C SC to further review the resolution of Recommendation No. 3.

5. Member Ballinger led a discussion of the ACRS activities associated with the increased enrichment draft rulemaking package to include:

[1] NUREG/CR-6303 was issued in 1994 and defines four digital I&C system "echelons of defense" that are derived from a plant defense-in-depth model which credits independent fuel cladding, reactor pressure vessel, and containment boundaries: (1) control system, (2) reactor trip system, (3) engineered safeguards features actuation system, and (4) monitoring and indication system. These four "echelons of defense" are the foundation for the diversity assessment methods described in the rest of NUREG/CR-6303.

a. November 19, 2024 (Completed), SC meeting on RG 1.183, revision 2, topics: pathway specific source term using MELCOR (Electric Power Research Institute Modular Accident Analysis Program runs), and updated to 2023 source term presentation,

b. December 17 and 18, 2024, SC meeting on draft rule language and package,

c. December 19, 2024, SC meeting on entire RG 1.183, revision 2,

d. January 2025 SC meeting on transition break size, Fuel Fragmentation, Redistribution, and Dispersal guidance documents, and

e. Letter report on draft rule language during February 2025 FC meeting.

The Committee discussed the available documents supporting this activity and the need to review the documents in a timely manner to facilitate production of the letter during the February 2025 FC meeting.

6. Vice Chair Halnon and Members Bier and Dimitrijevic led a discussion of the use of probabilistic assessment (PRA) and advanced reactor reviews.

   A planning meeting was held on November 14, 2024, to discuss possible PRA/Regulatory Policy SC meetings on the use of PRA analysis for new/advanced reactor reviews. NRC Staff and various stakeholders would be invited to present. Agenda items would likely include topics such as the following: absolute versus relative risk metrics, cliff edge analysis, PRA completeness, and staff/industry guidance documents under development. ACRS staff will coordinate with the appropriate staff in Office of Nuclear Reactor Regulation (NRR) and Office of Nuclear Regulatory Research (RES) to set up SC meetings in 2025. The expected outcome of these SC meetings is to understand approaches planned to be used by new reactor applicants, and that potential issues are resolved on a generic basis rather than in the context of any one applicant.

   It was agreed that Member Bier would take the lead on this topic with support from the ACRS staff.

7. The Chief of the ACRS's Technical Support Branch led a discussion of the proposed calendar from FC and SC meetings in calendar year 2026. The proposed calendar was sent to all Members, Consultants and staff previously. The Committee approved the calendar presented with one change to conduct the April 2026 FC meeting from April 8 through 10, instead of April 1 through 3, 2026.

8. Member Ballinger led a discussion of potential review of Westinghouse topical report (TR), WCAP-18773-P/NP, "Higher Enrichment for Westinghouse and Combustion Engineering Fuel Designs." This TR reviews Westinghouse's computer codes and fuel evaluation methods for their applicability to fuel with enrichments above their currently approved limitation of uranium-235 enrichment of 5 weight percent. The TR does not address higher burnup, which is expected to have a more significant impact on fuel evaluation methods. The specific evaluation methods include nuclear core design, fuel performance, thermal-hydraulic design, and Loss-of-Coolant Accident (LOCA) and non-LOCA analyses. It documents required changes and/or provides justification for applicability of existing NRC-approved methods. The WCAP describes an increase in

fuel enrichment will not change any fuel rod geometric parameters or characteristics, nor impact the fuel's microstructure and also documents that most of their fuel evaluations methods have no limitations or conditions related to fuel enrichment.

Increased fuel enrichment potentially increases initial core reactivity and local power peaking. Increased initial reactivity is the consequence of having more fissile material in the core. This impacts the core's available shutdown margin, particularly at beginning of life. Similarly, increases in local power peaking may be the outcome of greater disparity in fission activity within and between fuel assemblies. Both the reactivity and local power peaking challenges may be mitigated through core design, which will require careful optimization of fuel and poisons to balance the competing goals of minimizing power peaking and efficient burnup. Maintaining this balance is further complicated by burnup over a fuel cycle. Regarding LOCA and non-LOCA analyses specifically, both reactivity and power peaking are important phenomena influencing safety criteria; however, these phenomena are explicitly considered in the preparation of analyses through model input related to reactor kinetics and decay heat. Their LOCA models do require updates to address nuclear physics data, the neutron capture correction, the normalized fission interaction frequency and gamma energy redistribution, as these models explicitly consider fuel enrichment. In addition, cladding rupture is required to be precluded for higher enriched fuel design when extended decay heat curves are necessary.

While noting the modifications required for LOCA analyses, the WCAP supports the conclusion that Westinghouse's fuel codes and evaluation method remain applicable. This is presented through qualitative justifications that fuel enrichment has either a minimal or negligible impact on fuel, fuel rod, and fuel assembly failure mechanisms. Further Westinghouse makes no changes to safety criteria.

Collectively, the impact of fuel enrichment is shown to be more of a core design and operations challenge than a safety challenge, which is not an unexpected outcome. The TR Table 3.0-1, entitled, "High Energy Core Design Fuel Skeleton Impact Summary," provides an overview of the evaluations. As such, members Ballinger and Martin recommendation that the committee does not review this TR.

The Committee agreed with the recommendation not to review this document.

9. Member Ballinger led a discussion about proposed future interactions with representatives from the Electric Power Research Institute (EPRI) similar to what the Committee did in 2023 and 2024. The Committee agreed with the proposal with a focus on prioritizing topics to support ACRS reviews.

10. As the Chair of the Fuels, Materials, and Structures SC, Member Ballinger led a discussion of potential visits to fuel fabrication facilities in September 2025. The possible sites to visit are the Framatome fuel fabrication facility (light water reactor fuel performance presentation/tour) in Richland, WA, or to the BWXT fuel fabrication facility (TRISO fuel/tour) in Lynchburg, VA. After some discussion, it was agreed that the preferred site would be the BWXT facility. Member Martin stated that he would help contribute to the proposed agenda.

11. Members Roberts and Palmtag led a discussion about possible review of two TRs related to the eVinci design: EVR-LIC-RL-002-P, Revision 0, "Nuclear Design

Methodology Topical Report" (Proprietary) and EVR-LIC-RL-003-P, Revision 0 "Westinghouse TRISO Fuel Design Methodology Topical Report" (Proprietary).

Fuel design: Westinghouse is using advanced gas cooled reactor (AGR) TRISO particles qualified in accordance with the 2020 EPRI TR and its SE report, and the scope of the TR is limited to the TRISO particles. Given these bounds, member Roberts thought about not reviewing this at all. However, Westinghouse points out their particle size is larger than the AGR database. Given this distinction, member Roberts recommended that there would be value in the Committee, particularly Member-At-Large Petti, in pursuing an explanation of how larger particles fit within the AGR qualification box.

Nuclear Design: Part of this TR establishes adequacy of the validation basis for use of SERPENT to model the eVinci core. The first sentence of section 6.2 states, "Serpent has been validated for many different nuclear systems (e.g., References 57, 58, 59, and 60); however, none of those systems adequately capture the nuclear characteristics of the eVinci microreactor." Based on the importance of the validation basis for the nuclear design of a novel microreactor, ACRS review is recommended.

The Committee agreed with the recommendation to reviews these TRs.

12. Member Martin led a discussion about potential review of Global Nuclear Fuels - Americas (GNF-A) TR, NEDO 33935, Supplement 1, Revision 0, "Implementation of LANCR02/PANAC11 in Downstream Methods."

    In December 2021, GNF-A submitted three topical reports (TRs), including NEDO-33935P, Revision 0, "LANCR02/PANAC11 Application Methodology." That TR describes the integration of the LANCR02 lattice physics and PANAC11 core simulator to replace their previously approved TGBLA06/PANAC11 methodology. NRR met with ACRS in June 2023 to review the methodology and the staff SE, concluding that the update was sufficiently justified based on the use of vetted basic principles and extensive validation.

    Supplement 1 of NEDO-33935P was submitted by GNF-A in October 2022. It describes how LANCR02 will be implemented as a replacement for the current generation lattice physics code reactor pressure, and other design parameters. In NRR's draft SE, the staff evaluated the changes to downstream methods and concluded that they are minor.

    Member Martin reviewed the TR and the draft SE report. While the LANCR02/PANAC11 methodology itself introduces some computational and methodological advancements (e.g., enhanced lattice physics and pressure loss modeling), the implementation in downstream codes and methods is primarily administrative and procedural in nature. Member Martin recommends that the committee not review this TR.

    The Committee agreed with the recommendation not to review the document.

13. Member Palmtag led a discussion about potential review of PWROG-22021-P/NP, Revision 0, "Justifications for the Proposed Changes to the Quadrant Power Tilt Ratio Technical Specification."

    This document describes five proposed changes to the Quadrant Power Tilt Ratio (QPTR) Technical Specifications in NUREG-1431 "Standard Technical Specifications

Westinghouse Plants." The purpose of these changes is to improve operational flexibility. One important point to make is that the QPTR is not the same as the quadrant power tilt.

Briefly, the QPTR is the ratio of the maximum excore detector signal divided by the average detector signal at a certain elevation after the excore detectors have been calibrated. After calibration, QPTR is equal to 1.0. If QPTR increases, it indicates that asymmetries have been introduced into the core, which could be due to changing power shapes or by unsymmetric rod movements. Required actions are taken if QPTR exceeds 1.02.

The purpose of the QPTR limit is to protect the heat flux hot channel factor $F_Q(Z)$ and nuclear enthalpy rise hot channel factor $F^N_{\Delta H}$ limits, which are only "measured" at certain points of the operation." If QPTR is exceeded, it may indicate that the $F_Q(Z)$ and $F^N_{\Delta H}$ limits have been exceeded. The $F_Q(Z)$ and $F^N_{\Delta H}$ limits are in place to "ensure that fuel design limits for Departure from Nucleate Boiling and Peak Cladding Temperature will not be exceeded."

The changes to the TSs include:

1. Added a note so that the required actions are not required until the initial calibration of the excore channels is performed, which usually occurs during startup when 100% full power is reached. The issue is that during reactor startup, the calibration factors were obtained from the previous cycle, and these calibration factors may not be applicable to the current core conditions.

2. Changed the penalty function for the reactor power when QPTR exceeds 1.02. The current required action is to "Reduce THERMAL POWER > 3% from reactor thermal power for each 1% of QPTR > 1.00." The proposed change is to change the point 1.00 to 1.02. The document describes why the value of 1.00 is overly conservative and how uncertainty in quadrant power changes is already accounted for in the margin for uncertainty in the $F_Q(Z)$ and $F^N_{\Delta H}$ limits.

3. Allowed the QPTR limit to be exceed if there is sufficient margin in the $F_Q(Z)$ and $F^N_{\Delta H}$ limits. Often the core operates with sufficient margin in the $F_Q(Z)$ and $F^N_{\Delta H}$ limits, so that even if QPTR exceeds 1.02, the maximum $F_Q(Z)$ and $F^N_{\Delta H}$ limits would not be exceeded.

4. Combined two surveillance requirements to simplify the wording and cover the situations when the thermal power is less than and greater than 75% reactor thermal power.

5. Eliminated three required actions that are duplicated after making the above changes.

Member Palmtag reviewed the TR and recommends that the committee not review the report.

The Committee agreed with the recommendation not to review the document.

14. There were no additional reconciliations to discuss at this FC meeting.

15. Chair Kirchner led a discussion about review of the NuScale standard design approval application chapter memoranda for Chapters 3 (Halnon lead), 8 (Roberts), and 14 (Sunseri).

    Chapter 8 and 14 memoranda were discussed and finalized:

    1. For Chapter 8, it was agreed that Member Roberts would look at standard recommendation wording regarding that no further review of the chapter is necessary unless the review of Chapter 15 reveals an issue.

    2. The Chapter 3 memorandum was also discussed and after some discussion, it was agreed that there would be further interaction with NuScale and the staff on various topics at a future SC meeting.

16. Vice Chair Halnon led a discussion about a visit to Seabrook on April 17, 2025, for further review of the alkali-silica reaction. This may be combined with a visit to Westinghouse's Newington component manufacturing site.

    Members were asked to provide their input regarding intent to attend such a visit. Input should be provided to Vice Chair Halnon and the ACRS staff.

    Vice Chair Halnon stated that this visit would be undertaken in lieu of a visit to a Region, and he noted that coordination with Region I would be an integral part of the visit to Seabrook.

17. Member Roberts requested Committee approval for himself to represent ACRS at an upcoming American Nuclear Society (ANS) event.

    On December 3, 2024, ACRS staff received an email invitation from the NRC staff requesting Member Roberts speak at an ANS human factors and instrumentation and controls workshop prior to the Nuclear Plant Instrumentation and Control and Human-Machine Interface Technology meeting next year.

    The Committee approved Member Roberts attendance as a representative of the ACRS.

18. Under additional topics, Vice Chair Halnon stated that he would like to institute a regular review of guidance documents that have been published for the purpose of keeping the Members up to date on current guidance. He would do this during planning and procedure sessions in future FC meetings.

    On a different topic, Member Sunseri suggested consideration of a possible site visit to the Kairos Hermes facility that is under construction at Oak Ridge, TN.

19. Executive Director Moore led the conduct of annual ACRS officer elections in accordance with ACRS Bylaws, Chapter 8.

    The following were the results of the election:

Chair – Kirchner
Vice Chair – Halnon
Member-at-Large – Petti

20. The following topics are on the agenda of the 722nd ACRS Full Committee meeting, which will be held on February 5 through 7, 2025:

- Increased Enrichment Draft Rule Language and associated draft regulatory guides,

- Regulatory Guide 1.183, Revision 2, and

- NuScale loss-of-coolant accident methodology topical report.

Sincerely,

Walter L Kirchner Signed by Kirchner, Walter on 01/13/25

Walter L. Kirchner
Chair

Enclosure:

List of Acronyms

January 13, 2025

SUBJECT: SUMMARY REPORT – 721st MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, DECEMBER 4, 2024

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List of Acronyms

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents and Management System
AGR	Advanced Gas Cooled Reactor
ANS	American Nuclear Society
BTP	Branch Technical Position
CE	Combustion Engineering
EPRI	Electric Power Research Institute
FC	Full Committee
$F_Q(Z)$	Heat Flux Hot Channel Factor
$F^N_{\Delta H}$	Nuclear Enthalpy Rise Hot Channel Factor
GNF-A	Global Nuclear Fuels – Americas
I&C	Instrumentation and Control
LOCA	Loss-of-Coolant Accident
MRP	Material Reliability Program
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
RG	Regulatory Guide
PWR	Pressurized Water Reactor
PRA	Probabilistic Assessment
QPTR	Quadrant Power Tilt Ratio
RES	Office of Nuclear Regulatory Research
SC	Subcommittee
SE	Safety Evaluation
TR	Topical Report