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

December 4, 2023

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

SUBJECT: SUMMARY REPORT – 710th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, NOVEMBER 1-2, 2023

Dear Chair Hanson:

During its 710th meeting, November 1-2, 2023, which was conducted in person and virtually, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters. The ACRS completed the following correspondence:

LETTER

Letter to Daniel H. Dorman, Executive Director for Operations, U.S. Nuclear Regulatory Commission (NRC) from Joy L. Rempe, Chairman, ACRS:

- Interim Letter on Level 3 Probabilistic Risk Assessment – Volumes 3 and 4, Pertaining to Reactor at-Power Events, dated November 24, 2023, Agencywide Documents Access and Management System (ADAMS) Accession No. ML23317A199.

MEMORANDA

Memorandum to Daniel H. Dorman, 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 November 2023, dated November 8, 2023, ADAMS Accession No. ML23312A084.

- Regulatory Guides, dated November 8, 2023, ADAMS Accession No. ML23312A069.

HIGHLIGHTS OF KEY ISSUES

a. Interim Letter on Level 3 Probabilistic Risk Assessment (L3PRA) – Volumes 3 and 4, Pertaining to Reactor at-Power Events

The Committee heard from the NRC staff and issued a November 24, 2023, letter, with the following conclusions and recommendations:

1. When completed, the L3PRA study will be the most comprehensive full-scope probabilistic risk assessment (PRA) performed by NRC. The coverage of the PRA subject matter, including risks associated with severe accidents, is extensive. It applies experience gained over the 30 years since NUREG-1150, providing new insights related to regulatory decision-making and Level 3 PRA documentation, technical feasibility, and cost.

2. Insights, assumptions, sensitivity runs, treatment of uncertainties, model limitations, deficiencies, and possible enhancements are found throughout Volumes 3 and 4. The forthcoming Volume 1 summary report should cover the important insights from these volumes, including and expanding upon the items identified in this letter.

3. Given that results from this work could provide important risk insights for regulatory decision-making, resources should be prioritized to ensure that the remaining documents are issued without significant delays.

b. Regulatory Basis Document to Identify Regulatory Issues Associated with Increased Enrichment of Conventional and Accident Tolerant Fuel Deigns for Light Water Reactors

During the discussion on this topic, staff identified revisions needed to 10 CFR Part 50, 10 CFR 51.51, 10 CFR 51.52, 10 CFR 71.55, 10CFR 50.67/General Design Criterion 19 (GDC-19), and 10 CFR 50.46. Committee discussions primarily focused on proposed revisions to 10 CFR 50.67/GDC-19 [understanding the potential risk and safety significance of the proposed increase in allowable control room dose during accidents] and 10 CFR 50.46 [understanding the effects of fuel fragmentation, relocation, and dispersal (FFRD)]. With respect to the latter item, the staff presented options that ranged from no action to the use of probabilistic fracture mechanics, allowing one to conclude that leaks in large diameter pipes would be identified before failure, thereby precluding the need to analyze FFRD in large break Loss of Coolant Accidents (LOCAs). Because the stakeholder comment period has not ended, staff did not provide a recommended alternative for the item concerning FFRD at this meeting.

During the discussion, members offered a list of potential constraints and considerations that should be addressed as the FFRD options are evaluated:

- Calculations for a high burnup core (beyond the existing licensing basis of 62 GWd/MTU). These calculations suggest large amounts of fuel rod ballooning with existing fuel designs.

- Analyses of feasible extended burnup core and fuel designs. These should first demonstrate that fuel design limits required for both steady state operation and non-LOCA limiting transient events are maintained. These analyses would assure fuel management approaches and associated fuel duty for the double-ended guillotine break LOCA are not atypical.

- The limitations of the empirical FFRD database described in Regulatory Information Letter 213 (RIL 2021-13)[1] and the associated thresholds derived therein. These thresholds were used in a staff paper[2] to examine projected FFRD performance characteristics.

- The challenge to identify and express the uncertainties in the phenomena associated with FFRD and their quantification due to the multi-variate nature of the problem and non-prototypicality of some of the testing. These concerns were noted in our letter report on RIL 2021-13.[3]

- Analytic difficulty in developing defensible calculations for FFRD to address transport, coolability, re-criticality from the fuel debris, and the resultant impact on fuel performance and the radiological source term.

- Consideration of accident tolerant fuel and other advanced fuel/cladding designs that may limit FFRD occurrence or consequences.

- The frequency of a large break calculated by xLPR[4,5] (1E-07 per year). This frequency is well below the frequency range traditionally considered for design basis events (10E-2 to 1E-04 per year)[6], and it is consistent with the frequency estimated by a panel of experts in NUREG-1829.[7] Considering both the NUREG-1829 report and a draft rule for a risk-informed approach to large break LOCA, our letter (dated November 16, 2006)[8], concluded that the large break should still be considered as part of the safety analysis, albeit as a beyond design basis event for which some relaxation in design and analysis requirements could be justified, to provide sufficient defense-in-depth.

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[1] U.S. Nuclear Regulatory Commission (NRC), RIL 2021-13, “Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup,” December 2021 (ADAMS Accession No. ML21313A145).
[2] A. Bielen, J. Corson, and J. Staudenmeier, “NRC's Methodology to Estimate Fuel Dispersal during a Large Break Loss of Coolant Accident,” presented at the 20th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, April 2023 (Adams Accession No. ML23116A214).
[3] Advisory Committee on Reactor Safeguards, “Research Information Letter (RIL) 2021-13 on Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup,” December 20, 2021 (Adams Accession No. ML21347A940).
[4] NUREG-2247, “Extremely Low Probability of Rupture Version 2 Probabilistic Fracture Mechanics Code,” U.S. Nuclear Regulatory Commission, August 2021 (ADAMS Accession No. ML21225A736).
[5] N. Glunt, C, Harrington, and JD Shim (EPRI); M. Burkardt and G. Schmidt (DEI); “Use of the xLPR Code for Developing LOCA Frequency Estimates,” NRC Public Meeting, January 2023 (ADAMS Accession No. ML23019A162)
[6] Nuclear Energy Institute, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” NEI Technical Report 18-04, Rev 1, August 2019 (Adams Accession No. ML19241A472).
[7] NUREG-1829, Volume 1, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” U.S. Nuclear Regulatory Commission, April 2008 (Adams Accession No. ML082250436).
[8] As stated in “Draft Final Rule to Risk-Inform 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors” (Adams Accession No. ML063190465), “The Rule to risk-inform 10 CFR 50.46 should not be issued in its current form. It should be revised to strengthen the assurance of defense in depth for breaks beyond the transition break size (TBS). Such assurance would reduce concerns about uncertainties in determining the TBS.”

- Current assessment of the soundness of the underlying technical basis of probabilistic fracture mechanics and the associated uncertainties as used in xLPR calculations.

- The historic role that a double-ended guillotine break LOCA plays in the 50 years of nuclear safety regulation. It is a very strong precedent.

- The role of defense-in-depth to compensate for uncertainty in both frequency and consequences of FFRD.

- Adequate balance between defense-in-depth and risk significance to establish the proper (not excessive) safety margin for this issue in the context of a modern risk-informed regulator.

During the discussion, it was noted that several expected events may impact the path forward for addressing FFRD. These include:

- Detailed stakeholder comments on the regulatory basis will be received within the next few months.[9]

- A formal Phenomena Identification and Ranking Table (PIRT) process on this topic will be conducted in the spring of 2024.

- Industry (the Electric Power Research Institute [EPRI]) will be providing a formal submittal of what has been termed the "Alternative Licensing Strategy" in the first quarter of 2024.[10] This strategy uses details of the probabilistic fracture mechanics approach to address the FFRD issue.

- The stakeholder comments on the regulatory basis will be considered by the staff in 2024 to support a rulemaking.

For the above reasons, members concurred that a formal ACRS letter was not appropriate at this time.

c. <u>Discussions at the Planning and Procedures (P&P) Session</u>

1. The Committee discussed the Full Committee and Subcommittee schedules through April 2024 as well as the planned agenda items for Full Committee meetings. It was agreed that a special Full Committee meeting (scheduled on December 15, 2023) would only be held if letter writing was not completed during the regular December Full Committee meeting.

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[9] See Federal Register Notice, "Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors" at https://www.federalregister.gov/documents/2023/09/08/2023-19452/increased-enrichment-of-conventional-and-accident-tolerant-fuel-designs-for-light-water-reactors?utm_source=federalregister.gov&utm_medium=email&utm_campaign=subscription+mailing+list

[10] Letter from J. Uhle, NEI, to A. Veil, US NRC, "Industry Recommendations for a 10 CFR50.46a/c Combined Rulemaking," dated March 31, 2023 (Adams Accession No. ML23107A230).

2. The ACRS Executive Director led a discussion of significant notices issued by the Agency since the last Full Committee meeting in October 2023. The Executive Director documented this activity in a memorandum dated November 8, 2023, ADAMS Accession No. ML23312A084.

3. The ACRS Executive Director led a discussion of three draft regulatory guides/regulatory guides that the Committee was asked about regarding interest in review. The Executive Director documented the Committee's decision in a memorandum dated November 18, 2023, ADAMS Accession No. ML23312A069.

4. Members Brown and Roberts led a discussion on the proposal to not review the Kairos Instrument Setpoint Methodology topical report (TR), KP-NRC-2305-004, (ADAMS Accession No. ML23152A183). This report describes the methodology for establishing the Kairos Power Fluoride Salt-Cooled, High Temperature Reactors (KP-FHR) safety-related instrument setpoints. The methodology is consistent with American National Standards Institute (ANSI)/International Society of Automation (ISA) standard ANSI/ISA-67.04.01-2018, "Setpoints for Nuclear Safety-Related Instrumentation," requirements (Reference 1) as endorsed by Regulatory Guide 1.105, Revision 4, "Setpoints for Safety-Related Instrumentation."

   The Committee agreed with the recommendation.

5. Members Roberts and Brown led a discussion on the proposal to not review WCAP-18762-P, Revision 0, "Advanced Logic System® [ALS] v2 Platform Topical Report." WCAP-18762-P updates the TR for the "Advanced Logic System (ALS)" platform, previously approved by the NRC staff in a safety evaluation dated September 9, 2013 (ADAMS Accession No. ML13218A979), to reflect upgrades in building blocks, such as circuit card designs, used to develop an ALS-based instrumentation and control (I&C) system. This TR is not specific to the eVinci plant design but was submitted as part of the eVinci program to support use of the upgraded components in the eVinci design.

   The 2013 ALS TR was not reviewed by the Committee. Rather, the first integrated system that used the ALS building blocks, for the Diablo Canyon plant, was reviewed by the Committee in 2016 (ADAMS Accession No. ML16257A534). It is appropriate for the Committee to review an I&C system as an integrated system, to ensure that the I&C design principles are met. Conversely, it is not possible to determine if an integrated system using the ALS building blocks described in the 2013 TR, or updated in the referenced eVinci TR, will meet those overarching principles without an application. Therefore, there is no significant benefit to conducting an ACRS review on this TR. Rather, the eVinci integrated I&C system TR, scheduled for submittal in 2026, should be reviewed by both the eVinci and digital I&C SCs. Thus, Members Roberts and Brown recommended that the Committee not review this TR.

   The Committee agreed with the recommendation.

6. Members Roberts and Brown led a discussion on the proposal to not review WCAP-18780-P, Revision 0 "Advanced Logic System® v2 Development Process Topical Report." The referenced TR describes the processes that the applicant plans to follow when developing software for use in the Westinghouse "Advanced Logic System (ALS)" platform. The software development process is consistent with life-cycle processes such

as those defined by IEEE Standards 1074-2006, "IEEE Standard for Developing a Software Project Life Cycle Process," and 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations." It is not specific to the eVinci plant design but was submitted as part of the eVinci program to support use of the ALS components in the eVinci design.

While defined life-cycle processes are necessary to result in high quality software that supports safety objectives, the processes defined in the referenced TR are consistent with those used in any major software development. None of them are new or innovative. A Committee review is unlikely to add value after the NRC staff has completed their detailed review. Therefore, there is no significant benefit to conducting a review on this TR. Thus, Members Roberts and Brown recommended to not review this TR.

The Committee agreed with the recommendation.

7. Member-at-large Petti updated the Committee on the review of the Technology Inclusive Content of Application Project and Advanced Reactor Content of Application Project guidance documents. These documents are available for review. The Regulatory Policies and Practices – Part 53 Subcommittee plans to meet on November 16, 2023, to discuss this matter.

8. Member Halnon led a discussion on a potential visit to Region II in the Summer of 2024. Member Halnon mentioned visiting the Technical Training Center in Chattanooga to see how they are preparing for potential advanced reactor applications. There were discussions amongst the Committee about additional locations to visit such as the Kairos Hermes construction site and/or an operating plant in the area. It was noted that the Kairos 2 construction permit application review activities may be occurring at the same time and that the Region II visit would need to be arranged to reflect this higher priority activity.

9. Executive Director Moore led a discussion on the conduct of the Committee Officer elections that will be held during the December 2023 Full Committee meeting. He reviewed the applicable bylaws and highlighted that if any Member wanted to remove their names from consideration for the Chair or Vice Chair position, they should notify him via email by November 22, 2023.

10. Executive Director Moore led a discussion of a reconciliation on the topic of RG 1.183 Revision 1, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." During the discussion, members commented about the vagueness in the staff response, which did not elaborate upon which ACRS recommendations would be incorporated into Revision 2 of RG 1.183. Member-at-large Petti recommended no further action on this topic at this time, noting future reviews of Revision 2 of the regulatory guide would consider the adequacy of staff actions to address our comments.

    The Committee agreed with the recommendation.

11. There was a closed session of the P&P to discuss proprietary Committee Engagement Plans as well as sensitive administrative and personnel issues.

d. <u>Scheduled Topics for the 711th ACRS Meeting</u>

The following topics are on the agenda for the 711th ACRS meeting scheduled for December 6-8, 2023:

- Technology Inclusive Content of Application Project and Advanced Reactor Content of Application Project guidance documents, and
- Transportation Framework for Micro-reactors.

Note that a special Full Committee meeting is planned on December 15, 2023. This meeting will only be conducted if letter writing for these two topics is not completed during the regular December Full Committee meeting.

Sincerely,

Joy L. Rempe Signed by Rempe, Joy on 12/04/23

Joy L. Rempe
Chairman

December 4, 2023

SUBJECT: SUMMARY REPORT – 710th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, NOVEMBER 1-2, 2023

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