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

March 17, 2026

The Honorable Ho K. Nieh
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SUMMARY REPORT AND CERTIFIED MINUTES – 732ND MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, FEBRUARY 5, 2026

Dear Chairman Nieh:

During its 732nd meeting held February 5, 2026, which was conducted in person and virtually, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters. The Committee's efforts continued to be focused on statutorily required functions as directed in Executive Order (EO) 14300, "Ordering the Reform of the Nuclear Regulatory Commission."

This document serves as the certified minutes as required by the Federal Advisory Committee Act and Title 10 of the *Code of Federal Regulations* Part 7.

During this meeting the ACRS completed the following correspondence:

MEMORANDA

Memoranda to Michael F. King, Executive Director for Operations, U.S. Nuclear Regulatory Commission (NRC), from Marissa G. Bailey, Executive Director, ACRS:

- February 2026 ACRS Full Committee Meeting – Topical Reports (TR), dated February 25, 2026, Agencywide Documents Access and Management System (ADAMS) Accession No. ML26040A091, and

- Documentation of Receipt of Applicable Official NRC Notices to the ACRS for February 2026, dated February 10, 2026, ADAMS Accession No. ML26040A145.

HIGHLIGHTS OF KEY ISSUES

The Agenda for this meeting was published in the *Federal Register* on January 14, 2026. The transcripts of the meeting are available on the ACRS's public website. The attendance list may be found in Enclosure 2 of this document.

A. Self-Assessment, Lessons Learned, and Path Forward for New Reactor Reviews

The Committee discussed the lessons learned from recent reviews of new reactor applications including the NuScale standard design approval application, the Terrapower Natrium reactor at the Kemmerer Power Station construction permit application (CPA), the Kairos Hermes 1 and 2 CPAs and SHINE Technology applications. Also discussed were the current review plans for the Tennessee Valley Authority's (TVA) Clinch River and X-energy Long Mott CPA. The Committee discussed the reviews in light of the requirements of the Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy (ADVANCE) Act and EO 14300, "Ordering the Reform on the Nuclear Regulatory Commission." Vice Chairman Petti led this discussion and read in for the record a proposed letter report that would capture the Committee's thoughts. The Committee will continue discussing this issue at the March 2026 Full Committee (FC) meeting with a goal of finalizing the letter at that meeting.

B. Discussions during the Planning and Procedures (P&P) Session

1. The Committee discussed the FC and Subcommittee (SC) schedules through July 2026 as well as the planned agenda items for FC meetings.

2. There were no regulatory guides discussed at this meeting.

3. The Committee heard from the SC Chairman (Member-at-Large Harrington) regarding a SC engagement that was conducted on February 4, 2026:

    a. New Reactor Subcommittee closed engagement on the Clinch River CPA, which discussed the unique, novel, and noteworthy aspects of the application and the overall review plan.

4. The Committee discussed the following ongoing and near-term new reactor application reviews:

    a. Long Mott Generating Station CPA. Member Martin led this discussion and reminded the members that a closed SC engagement was scheduled for February 17, 2026.

    Members were reminded that the review activities include an evaluation of the preliminary safety analysis report (PSAR) for unique, novel, and noteworthy (UNN) content and topic-level questions. These were provided to the Office of Nuclear Reactor Regulation (NRR) staff prior to the December 2025 FC meeting. More recently, a draft PSAR/UNN briefing presentation was prepared for the closed February 17, 2026, SC engagement with NRR staff. The presentation will step through Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 CPA/PSAR, highlighting the UNN topics. The applicant is applying a risk-informed Licensing Modernization Project approach based on Regulatory Guide (RG) 1.233, i.e., Nuclear Energy Institute (NEI) 18-04, supported by applicable non-light water reactor (LWR) probabilistic risk assessment standards and NRC guidance. It will also provide a chapter-level orientation with UNN focused summaries and a candidate list of lead member review assignments to enable a structured discussion.

    The advanced safety evaluation is scheduled to be sent to the ACRS in August 2026, and an ACRS letter report is planned soon after.

b. Clinch River CPA application. Member-at-Large Harrington continued his discussion of this review. The Committee agreed that he and the ACRS staff would further discuss the possibility of inviting the applicant (TVA) to a public SC meeting in March or April 2026 to discuss UNN topics along with the safety strategy and stability topical reports (TRs). It was also discussed that the NRR staff would be invited to participate and/or observe.

c. The Committee discussed briefly the following other ongoing and near-term new reactor applications: the Fermi America combined license application for four Westinghouse Advanced Passive 1000 reactors in Amarillo, Texas, the recently submitted Belews Creek early site permit application in Stokes County, North Carolina, and the Palisades limited work authorization application for the Holtec SMR-300 reactor in South Haven, Michigan.

5. The Committee discussed the status of high priority rulemaking activities including:

a. 10 CFR Part 57, "Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles." Vice Chairman Petti reiterated that the Committee plans to have a FC meeting on this topic and write a letter after the rule is published in the *Federal Register* for public comment.

b. Executive Order 14300 Rulemakings. Vice Chairman Petti and Technical Assistant Rob Krsek led a discussion on this topic. Subcommittee Engagements are being scheduled for Section 5 of EO 14300 and the Committee will decide which areas to review in Sections 5(b), 5(d), 5(e), 5(f), 5(g), and 5(h). If necessary, FC meetings and letter reports will be scheduled when the draft proposed rules are published in the *Federal Register* for public comment.

6. The ACRS Executive Director led a discussion on the following TR review recommendations and documented the Committee's decision in a memorandum dated February 25, 2026. These TRs support new reactor facility applications and proposed reactor safety standards for operating reactors:

a. Holtec SMR-300 Time-Domain LS-DYNA Soil Structure Interaction Analysis Methodology Licensing TR (ADAMS Accession No. ML25330A200)

On November 26, 2025, SMR, LLC (Holtec) submitted the TR titled, "SMR-300 Time-Domain LS-DYNA Soil Structure Interaction Analysis Methodology Licensing Topical Report." Additionally, reports entitled, "Validation of a Simplified Fluid Structure Interaction Modeling Approach Used for the Seismic System Analysis of SMR-300," and "Validation of LS-DYNA for Soil Structure Interaction Analysis," were provided as supporting information.

The purpose of this submittal is to request that the NRC review and approve the soil-structure-interaction (SSI) methodology described in the report as technically acceptable and consistent with current regulations. The report describes a fully-coupled methodology using the time-domain seismic SSI modeling tool LS-DYNA. LS-DYNA is a commonly used and well-validated commercial code appropriate for an SSI analyses of the type described in the TR as per an evaluation in NUREG/CR-7193. The methodology also incorporates the modeling of fluid-structure interaction (FSI) effects using a simplified Lagrangian approach described in the applicable supporting validation report. The methodology incorporates three component ground motion loads as input, consistent with the Standard Review Plan (SRP) NUREG-0800. The three directions of input motion are used simultaneously in the SSI analysis. The calculated seven acceleration response spectra are then averaged for each location and each direction.

The methodology described in the report is site-independent and adheres to NUREG-0800, Section 3.7.2 and American Society of Concrete Engineers (ASCE) 4-16. The methodology is intended to be applied to demonstrate that reactor designs conform to NRC requirements in 10 CFR Part 50, Appendix A (General Design Control (GDC) 2) and Appendix S, as well as Standard Review Plan Section 3.7 and applicable ASCE/SEI standards.

Time-domain SSI analyses are a more recent analytical approach than the frequency-domain SSI analyses historically used for assessment of in-structure response spectra for Seismic Category I and Seismic Category II structures. Although newer, time-domain SSI analyses are more appropriate for complex SSI analyses, particularly when modeling embedded structures and non-linear aspects, such as the behavior of soils (as opposed rock materials) and fluids. The use of time-domain modeling approaches for deeply embedded foundations has been recognized as an appropriate approach since at least 2006, with the publication of NUREG/CR-6896. Several more recent publications provide guidance as to the appropriate application of tools such as described in the TR (e.g., DC/COL-ISG-017, RG 1.122, and ASCE/SEI 43-19).

Member Kammerer recommended the Committee not review this TR since the methodologies were not UNN nor safety significant. The Committee agreed with the recommendation.

b. Holtec SMR-300 Structural Modularity TR – Concrete Strengthened Steel Structures Design and Analysis Methodology (ADAMS Accession No. ML25336A060)

Holtec SMR, LLC, plans to construct certain SMR-300 structures using Concrete-Strengthened Steel Modules (CSSMs), a proven class of a steel-concrete composite design which is evolving in novel ways for safety-related nuclear applications. The CSSMs combine steel faceplates, stiffeners and cast-in-place concrete infill within a modular framework that enables accelerated construction and delivers robust structural performance. CSSMs are interconnected to form various building structural members such as structural walls, slabs, basemats, beams, columns and beam-columns. Regulatory Guide 1.243, “Safety-Related Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments,” provides substantial guidance for walls but does not extend into all areas of structural design desired for modularity economies. The RG also endorses American Institute for Steel Construction (AISC) N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities.” Nevertheless, important gaps remain in current industry codes and standards to comprehensively address the design or analysis of such structural elements; therefore, this TR establishes the design basis, analysis methodology, and acceptance criteria for the range of CSSM structural elements intended for use in the SMR-300. The TR consolidates and adapts applicable provisions from existing industry design codes and standards to provide a comprehensive methodology commensurate with regulatory requirements for safety-related and containment pressure retaining nuclear structures. Additionally, this TR includes provisions and design principles appropriate for quality assurance and control during fabrication and construction, as well as for lifecycle in-service inspection activities, ensuring continued structural reliability and safety throughout the plant's operational life.

The Committee has previously reviewed different, though largely similar, design and analysis methodologies described in previous TRs for a range of steel-concrete structural configurations for the AP1000, the NuScale SMR, and the BWRX-300. In each, slightly different approaches to faceplate attachment, anchoring to concrete, anchor element design and configuration, and treatment of interfacial sheer forces were used. Specifically, the method of connecting the parallel steel plates and anchoring the faceplates within the infill were the primary differences and were usually held in proprietary status. None of the proposed methods are an exotic application of material science and the associated design calculations are straight forward and employ widely-used methods. The Committee had no comments on previous applications of the steel concrete designs in either the associated TRs or the SERs. It is important to note that industry standards in this topical area continue to evolve. The latest revision of AISC N690 (i.e., AISC N690-24), which is invoked by this TR, expands its applicability in the use of steel concrete walls to a broader range of structural elements that may be desired for use in modular construction. The AISC N690-24 Standard is currently under review by the staff.

This TR is judged to not reflect a consequential departure from the range of steel-composite design configurations, design approaches, and analysis methodologies described within the several parallel TRs the Committee has previously reviewed. Furthermore, within this topical area the staff is involved in industry standards activity, is reviewing newly released standards for regulatory approval, and has proven proficient in determining TR technical acceptability. Therefore, Members Annie Kammerer and Gregory Halnon recommended ACRS not review the subject SMR-300 TR since the methodologies were not UNN nor safety significant. The Committee agreed to not review this document.

c. Revised Recommendation on Westinghouse TR, WCAP-16747 Appendices C and D, "POLCA-T: System Analysis Code with Three-Dimensional Core Model," Revision 2, (ADAMS Accession No. ML24267A196)

Note that at the September 2025 FC meeting, Member Martin made a recommendation to review this TR (and the Committee concurred). Since then, additional information has become available to the Committee. Below is a revised recommendation.

In December 2023, Westinghouse submitted TR WCAP-16747 Appendix C and D, Revision 2, "POLCA-T: System Analysis Code with Three-Dimensional Core Model," with application to Anticipated Operational Occurrences (AOOs) and Anticipated Transients Without Scram (ATWS) for Boiling Water Reactor (BWR)/2 through BWR/6 plants. As originally submitted in 2010, WCAP-16747 consolidated and modernized previously approved evaluation models based on Westinghouse-developed BWR safety codes, including GOBLIN, BISON, and

RIGEL, into POLCA-T. The POLCA-T based evaluation model inherited much of the earlier multi-physics modeling and closure relations while upgrading the core model to three-dimensions (3D). The WCAP-16747 with Appendices A (control rod drop accidents) and B (stability) was previously approved in 2018. Appendices C and D of this TR are drawn largely from the previously NRC-approved methodology in WCAP-17203-P-A, which defines the best-estimate plus uncertainty framework, PIRT basis, and statistical treatment of uncertainty (95/95) for fast transient AOOs and ATWS.

The primary innovation introduced in WCAP-16747 is the application of this methodology within a fully integrated 3D spatial kinetics code (POLCA-T). This enables higher-fidelity modeling of core power distribution and local reactivity feedback during transients, offering potential benefits for realism but also introducing new complexities. Of note is the application of 3D spatial kinetics in the treatment of fast transients and ATWS scenarios. In these cases, the evaluation model integrates high-fidelity multiphysics simulation within a best-estimate framework.
The methodology permits limited fuel cladding failure, provided that long-term core coolability is preserved in accordance with 10 CFR 50.62 and GDC 35. While this is permissible under current regulatory criteria, it represents a shift from prior evaluation models that typically assume fuel integrity as a bounding design objective.

The allowance for fuel failure is especially noteworthy considering the increased likelihood of fuel fragmentation, relocation, and dispersal (FFRD) under the high burnup and thermal loads typical of power uprate conditions. The POLCA-T methodology does not include modeling of post-failure fuel behavior, yet the analysis presumes that such failure does not compromise cooling or containment performance. This raises questions about the sufficiency of the safety demonstration and the basis for concluding that ATWS risk is acceptably mitigated without an intact fuel envelope.

On November 21, 2025, Members Martin, Petti, Halnon, and ACRS staff met with NRR staff to discuss the staff's review of WCAP-16747, Revision 2, and how FFRD considerations interface with the POLCA-T evaluation model. The NRC staff acknowledged the Committee's concerns, clarified that FFRD was not an explicit focus of the POLCA-T TR review, and indicated that they are considering how best to characterize the treatment, or non-treatment, of FFRD in the associated safety evaluation.

Based on this discussion, Member Martin recommended ACRS not review this TR since the methodologies were not UNN nor safety significant. Instead, any questions about FFRD in the context of POLCA-T (and similar evaluation models from other applicants) can be addressed, as appropriate, in broader discussions of FFRD in light-water reactor transient and accident analyses that are anticipated with the ongoing increased enrichment rulemaking. The Committee agreed with this recommendation.

d. Westinghouse WCAP-18483-P, "EnCore® Chromium Coated Cladding for Use in Pressurized Water Reactors," Revision 0, (ADAMS Accession No. ML25184A401)

The committee has been following developments in LWR “advanced” or “accident-tolerant” fuel cladding, including the benefits associated with chrome-coated zircalloy, which is being pursued in varying forms by several fuel vendors. This TR addresses specific aspects of the EnCore® Chromium Coated Cladding product, which includes a metallic chromium (Cr) coating applied to the outside surface of licensed zirconium alloy cladding. This product is designed to reduce cladding oxidation and wear under normal operating conditions, and to reduce oxidation under high temperature transient or accident conditions. This TR describes the coated cladding material properties and performance from a broad test program including laboratory testing and lead test rods in commercial reactors. The report addresses specific regulatory criteria and discusses incorporation into existing NRC-approved analytical methods for use in plant-specific safety analyses to support product implementation in operating commercial PWRs while ensuring regulatory compliance and safety.

As the initial report on a coated LWR fuel cladding product presented to ACRS for review, this TR is UNN and potentially safety significant. Much of the relevant technical content is proprietary, limiting ACRS’s review to closed sessions. Member Harrington recommended the committee formally review this TR. The Committee agreed with this recommendation.

e. Westinghouse WCAP-18951–P/NP, “TRITON11 Reference Fuel Design,” Revision 1, (ADAMS Accession No. ML25149A004)

In May 2025, Westinghouse Electric Company (Westinghouse) submitted TR, WCAP-18951-P/NP, Revision 1, “TRITON11 Reference Fuel Design,” to the NRC for review and approval (ADAMS Accession No. ML25149A003). This TR includes a description of the TRITON11 BWR fuel assembly design and describes the application of previously reviewed methods and methodologies to this design. The TRITON11 BWR fuel design is an evolutionary design built off the SVEA96 experience. Some of the new features of the TRITON11 design include an 11x11 array of fuel rods and three water tube rods, which replace the water cross used in previous SVEA96 designs.

This TR is similar to previous submittals of new BWR fuel designs and the methodologies were not UNN nor safety significant. The only potential UNN is the fuel burnup limits; however, this specific topic will be covered in other ACRS reviews.

Member Palmtag recommended that the committee not review this TR. The Committee agreed with this recommendation.

f. Framatome EMF-93-177, Supplement 3P, “Mechanical Design for BWR Fuel Channels: ATRIUM BWR Enhanced Accident Model,” Revision 0, and Supplement 3NP, “Mechanical Design for BWR Fuel Channels: ATRIUM BWR Enhanced Accident Model,” Revision 1, (ADAMS Accession Package No. ML25136A387)

Framatome’s approved Mechanical Design for BWR Fuel Channels TR establishes the licensing methodology for demonstrating that channeled BWR fuel assemblies remain structurally adequate under externally applied forces (notably the safe-shutdown earthquake and postulated pipe breaks/Loss Of Coolant Accident (LOCA)) in a manner consistent with NUREG-0800, SRP Section 4.2, Appendix A guidance. The methodology includes square-root-sum-of-the-squares combination of seismic and LOCA load contributions, consistent with 10 CFR 50, Appendix A, GDC 2 load-combination expectations.

Supplement 3 (A-BEAM) leverages the December 2023 update to Regulatory Guide 1.61, Revision 2, as a regulatory anchor to introduce an optional, more detailed horizontal dynamic accident model that: (1) reflects plant/industry evolution (e.g., explicit accommodation of plant-specific acceleration time histories at the core support and upper end guide), and (2) bundles broader model refinements and testing/verification. The supplement remains aligned with the SRP 4.2 Appendix A regulatory construct for earthquakes and postulated pipe breaks and preserves continuity with the base TR's validation philosophy. The result is a more refined and defensible estimate of accident-induced dynamic response for fuel channel components.

On validation and verification, instrumented fuel-channel testing and comparisons to analysis are presented. Supplement 3 extends applicable test verifications to support the updated model's behavior and includes full-bundle test program elements as part of its technical basis for the refined accident model.

Member Martin's initial assessment was that Supplement 3 is potentially noteworthy for ACRS review purposes. The supplement does not introduce new physics, new acceptance criteria, or a fundamentally new analytical paradigm; however, it formalizes a shift in accident analysis implementation from reliance on generic/bounding excitation inputs toward explicit use of plant-specific load inputs, particularly acceleration time histories applied at the core support and upper end guide. From an UNN perspective, the potential noteworthiness arises because this single implementation degree of freedom and generic versus plant-specific load definition, should be demonstrated to preserve a deterministic, conservative envelope for the governing channel-response figure of merit. Conversely, if the applicant's submittal and NRC staff review demonstrates that plant-specific load inputs are selected and applied in a manner that remains bounded and conservative (with clear envelopment/bounding rules and sensitivity results confirming that acceptability does not depend on discretionary input choices), then it would be appropriate to conclude that the supplement is not noteworthy.

The Committee discussed several aspects of this TR including the option to review the implementation of this TR during plant-specific applications. The Committee agreed that the lead members and ACRS staff would seek additional information from the NRR staff and that this topic will be re-discussed at the March or April FC meeting.

The Committee agreed that the lead members would provide the NRC staff with specific questions regarding the potential UNN aspects of the TR, the reason for the new methodology and any impacts on reductions to safety margin. The Committee will discuss this TR at a future FC meeting when additional information is obtained, likely in March 2026.

7. Member Roberts led a discussion about NRC staff safety evaluations that were recently issued for the Westinghouse eVinci design. The NRC staff recently informed the ACRS staff that they had issued safety evaluations for the following two TRs:

EVR-LIC-RL-002-P, "Nuclear Design Methodology Topical Report," (Proprietary) and EVR-LIC-RL-003-P, "Westinghouse TRISO Fuel Design Methodology Topical Report," (Proprietary). In the December 2024 ACRS FC meeting, as documented in the meeting summary report (ADAMS Accession No. ML24353A243), the ACRS determined that these two TRs should be reviewed by the Committee. This did not occur prior to the issuance of the NRC staff safety evaluation report. While the subsequent issuance of EO 14300 and associated guidance to staff in June 2025 resulted in the ACRS focusing its reviews on UNN topics and safety significant topics these two TRs were not removed from the ACRS review list. This summary serves as the Committee's closure of these two TRs.

Review of the final SEs by ACRS members Roberts, Palmtag, and Petti indicated that the SEs include limitations and conditions that defer resolution of the technical questions that prompted the Committee to select these TRs for review. Specifically:

- The nuclear design methodology TR was selected for review to ensure adequate validation and verification to apply the pre-existing SERPENT methodology to the specifics of the eVinci core design. This assessment is deferred to subsequent licensing documents, as stated in the following (non-proprietary) limitation, "*The TR describes analytical methodologies that are still under development. Therefore, the NRC staff's approval of the methods in this TR is limited to the general use of Serpent and the described model assumptions. The acceptability of the implementation of specific models or calculational techniques, model results, and their use to demonstrate conformance with regulatory requirements for the construction and operation of a facility, including PDCs [Principal Design Criteria], will be the subject of future reviews or license applications.*"

- The fuel design methodology TR was selected primarily to ensure that departures from the existing advanced gas reactor qualification basis for TRISO fuel were adequately justified. This assessment is also deferred, as stated in the following (non-proprietary) condition, "*An applicant referencing this TR must provide justification that parameters affecting the fuel performance, including the composition, dimensions, and operating envelope, are within the scope of those to which the methodology can be applied and its results validated.*"

Therefore, the questions that led the Committee to request review of these two TRs in December 2024 were addressed in the NRC staff's safety evaluations through limitations and conditions, which essentially defer resolution of the UNN technical topics into subsequent license applications. Therefore, the Committee does not have a safety concern with the evaluations issued by NRC staff for these two TRs and will plan to review the implementation of the limitations and conditions in subsequent applications for the eVinci reactor.

8. Member-at-Large Harrington led a discussion on a proposed wholesale revision of the Committee's Bylaws. A revised version was sent to all Members prior to the meeting and there was a discussion of the changes. The Members agreed they would provide input after this meeting and a final draft version would be sent, so that the Committee could vote on the revised Bylaws during the March 2026 FC meeting.

9. There were no reconciliations of ACRS letter reports discussed during this meeting.

10. The Executive Director of ACRS led a discussion of important NRC announcements ACRS members were required to review, which is documented in a memorandum to the Executive Director for Operations in, “Documentation of Receipt of Applicable Official NRC Notices to the ACRS for February 2026," dated February 10, 2026, (ADAMS Accession No. ML26040A145).

11. There was no closed session as part of this P&P.

12. The following topics are on the agenda for the 733rd ACRS FC meeting, which will be held March 5 through 6, 2026: Continued discussion of self-assessment, lessons learned, and path forward for new reactor application reviews.

I hereby certify, to the best of my knowledge and belief, that the above minutes of the subject meeting are an accurate record of the proceedings for that meeting. In accordance with 10 CFR Part 7, “ADVISORY COMMITTEES,” Section 7.13, “Minutes of advisory committee meetings.”

Sincerely,

Signed by Halnon, Gregory
on 03/17/26

Gregory H. Halnon
Chairman

Enclosures:

1.) List of Acronyms
2.) List of Online Attendees

LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
3D	Three-Dimensions
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
ADVANCE	Accelerating Deployment of Versatile, Advance Nuclear for Clean Energy
AISC	American Institute for Steel Construction
AOOs	Anticipated Operational Occurrences
ASCE	American Society of Civil Engineers
ATWS	Anticipated Transient Without Scram
BWR	Boiling-Water Reactor
CPA	Construction Permit Application
COL	Combined License
DC	Design Certification
EO	Executive Order
FC	Full Committee
FFRD	Fuel Fragmentation, redistribution, and dispersal
GDC	General Design Criterion
ISG	Interim Staff Guidance
NEI	Nuclear Energy Institute
LOCA	Loss of Coolant Accident
LWR	Light-water Reactor
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PDC	Principal Design Criteria
PSAR	Preliminary Safety Analysis Report
P&P	Planning and Procedures
RG	Regulatory Guide
SC	Subcommittee
SMR	Small Modular Reactor
SRP	Standard Review Plan
TR	Topical Report
TVA	Tennessee Valley Authority
UNN	Unique, Novel, or Noteworthy

ATTENDEES

Attendees	**Attendee Affiliation (If known)**
Annie Kammerer	ACRS Member
Craig Harrington	ACRS Member-At-Large
Dave Petti	ACRS Vice Chairman
Gregory Halnon	ACRS Chairman
Matt Sunseri (virtually)	ACRS Member
Scott Palmtag	ACRS Member
Thomas Roberts	ACRS Member
Walt Kirchner	ACRS Member
Robert Martin	ACRS Member
Alesha Bellinger	ACRS Staff
Andrea Torres	ACRS Staff
Christina Antonescu	ACRS Staff
Christopher Brown	ACRS Staff
Janet Riner	ACRS Staff
Kent Howard	ACRS Staff
Larry Burkhart	ACRS Staff
Marissa Bailey	ACRS Staff
Quynh Nguyen	ACRS Staff
Rob Krsek	ACRS Staff
Sandra Walker	ACRS Staff
Shandeth Walton	ACRS Staff
Tammy Skov	ACRS Staff
Weidong Wang	ACRS Staff
Adrian Muniz	NRC Staff
Angel Moreno	NRC Staff
Carol Dye	NRC Staff
Denise McGovern	NRC Staff
India Banks	NRC Staff
Michelle Hayes	NRC Staff
Ryan Nolan	NRC Staff
Andrew Mauer	Nuclear Energy Institute
Antonio Godoy	
Suresh Kalyanam	Batelle Memorial Institute
Sarah Lopas	Terrapower
Andrew Zach	Senate Environment and Public Works Committee