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

May 16, 2025

The Honorable David A. Wright
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SUMMARY REPORT – 724th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, APRIL 2-3, 2025

Dear Chairman Wright:

During its 724th meeting held April 2 through 3, 2025, 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 Dr. Mirela Gavrilas, Executive Director for Operations, U.S. Nuclear Regulatory Commission (NRC), from Walter L. Kirchner, Chairman, ACRS:

- "Terrestrial Energy's Principal Design Criteria for the Integrated Molten Salt Reactor," dated April 21, 2025, Agencywide Documents Access and Management System (ADAMS) Accession No ML25099A144.

MEMORANDA

Memoranda to Dr. Mirela Gavrilas, Executive Director for Operations, U.S. Nuclear Regulatory Commission (NRC), from Marissa G. Bailey, Executive Director, ACRS:

- Documentation of Receipt of Applicable Official NRC Notices to the Advisory Committee on Reactor Safeguards for April 2025, dated April 9, 2025, ADAMS Accession No. ML25099A122,

- April 2025 Advisory Committee on Reactor Safeguards (ACRS) Full Committee – Topical Reports, dated April 9, 2025, ADAMS Accession No. ML25099A129, and

- Regulatory Guides (RGs), dated April 9, 2025, ADAMS Accession No. ML25099A117.

HIGHLIGHTS OF KEY ISSUES

a. Terrestrial Energy Topical Report (TR) on Principal Design Criteria (PDC) on the Integrated Molten Salt Reactor (IMSR)

The Committee met with representatives from Terrestrial Energy USA, Inc. (TEUSA) and the NRC staff on this topic and issued its letter dated April 21, 2025, with the following recommendations and conclusions:

1. The PDC proposed by TEUSA for the IMSR reactor have been developed by adapting Advanced (Non-Light Water) Reactor design criteria from NRC guidance; design criteria from a draft guidance in the American National Standards Institute (ANSI)/American Nuclear Society (ANS) ANSI/ANS-20.2-2023, “Nuclear Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten Salt Reactor (MSR) Nuclear Power Plants”; and consideration of the unique design features of the IMSR.

2. The use of a negative fuel salt temperature coefficient as the sole means of placing and maintaining the reactor in a “safe state” has not yet been demonstrated for this design. Additionally, the use of a “safe state” as equivalent to “safe shutdown,” with long-term criticality as an acceptable post-accident state, is a significant departure from accepted nuclear safety practices. The following have not been justified for this first-of-a-kind reactor:

    a. Absence of an automatic reactor protection system to ensure that the reactor can always be placed in a safe condition.

    b. Lack of a shutdown system with appropriate margin for malfunctions to ensure, that post accident, the reactor can be maintained in a subcritical state, not just a “safe state.” This position is consistent with the ANSI/ANS MSR Standard Criteria 20, “Protection System Functions,” and 26, “Reactivity Control and Redundancy.”

3. The PDC proposed by TEUSA remove the requirement for a containment cleanup system as found in Standard Criterion 41, “Containment Atmosphere Cleanup,” of the draft ANSI/ANS standard. The Committee considers this premature given that the final design is not complete.

4. The PDC are foundational to the overall safe design of the reactor. Therefore, they should be available in a non-proprietary format to provide transparency to the public.

5. The staff should consider these comments prior to issuing the final safety evaluation.

b. NuScale Standard Design Approval Application (SDAA) Topics Including NuScale TRs on Extended Passive Cooling and Reactivity Control Methodology and Non-Loss of Coolant Accident (Non-LOCA) Methodology

The Committee met with NuScale representatives and the NRC staff on the two TRs and other related topics. The Committee also discussed topics in preparation for drafting the Committee's final letter report on the SDAA at the May 2025 full committee meeting.

Member Martin reviewed NuScale's TR-0516-49416, "Non-Loss-of-Coolant Accident Analysis Methodology," Revision 4 (ADAMS Accession No. ML23005A305), describing the non-LOCA evaluation model (EM) for design-basis transient analyses in the 250 megawatt thermal (MWth) NuScale Power Module (NPM-20). The Committee reviewed a previous version of this TR in 2020 for use with the 160 MWth NPM-160, providing a letter report at that time (ADAMS Accession No. ML20085K048). Revision 4 updates the model to support the uprated US460 design.

On March 4, 2025, NuScale and NRC staff presented the revised TR and supporting analyses to the Committee. The non-LOCA EM follows established regulatory guidance, including RG 1.203 and the NuScale Design-Specific Review Standard, and retains key elements of the previously approved methodology, including event classification, system response analyses, and demonstration of fuel and radiological safety criteria without operator action for 72 hours. The applicant affirmed that the methodology identifies limiting single failures, accounts for the potential negative influence of non-safety system actions and includes bounding assumptions as appropriate.

The NRC staff's review concluded that the revised EM supports the finding of reasonable assurance of safety, subject to 10 Limitations and Conditions (L&Cs). Most L&Cs are consistent with those applied to the previously approved methodology; however, several were updated to reflect changes in the NPM-20 design and modeling tools. Among these, L&C No. 4, requiring a bias on decay heat removal system (DHRS) heat transfer, was a focal point of discussion during the subcommittee meeting. The staff cited concerns related to scaling and model uncertainty as justification for the application of a bias, despite NuScale's presentation of test data and analyses intended to support the adequacy of the realistic DHRS model.

The Committee concludes that NuScale's revised non-LOCA EM remains technically sound and sufficiently conservative for evaluating the NPM-20's response to design-basis transients. This conclusion is supported primarily by its continuity with an already approved methodology and a reaffirmed focus on dominant phenomena and critical figures-of-merit (FOM).

Regarding the staff's safety evaluation (SE), the Committee has no objection to its issuance; however, the Committee observes that L&C No. 4 does not constitute a safety issue, and its removal would be consistent with the technical basis presented. The application of a bias on DHRS heat transfer is unwarranted, as the underlying uncertainty relates to standard design considerations, not unmodeled phenomena or scaling distortion. The steam generator/DHRS configuration reflects well-understood industrial heat exchange principles, where sufficiently sized heat transfer surface area ensures heat rejection with minimal sensitivity to uncertainties. Given NuScale's new test results and modeling showing that the system maintains ample margin to avoid overpressure, the bias unnecessarily duplicates conservatism inherent in the system design and undermines the credibility of NuScale's validated approach.

It is recommended that this write-up serves as the record of the subcommittee meeting and that an ACRS letter report not be prepared. The Committee agreed with this summary.

Regarding the NuScale's TR-124587, "Extended Passive Cooling and Reactivity Control Methodology," Member Palmtag led the review.

On March 4, 2025, the NuScale subcommittee of the ACRS reviewed the NuScale TR "Extended Passive Cooling and Reactivity Control Methodology," Revision 0. This TR describes the methodology to evaluate the emergency core cooling system (ECCS) and DHRS extended passive cooling (XPC) function. The report is applicable to both loss-of-coolant accident (LOCA) and non-LOCA design basis events and shows compliance with regulatory requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46(b)(4) for long term cooling, and 10 CFR 50.46(b)(5) for coolable geometry. The report also shows compliance with General Design Criteria (GDC) 26, 27, 34, and 35.

In the XPC LR, NuScale presents the FOM selected for the XPC EM. These include: a) subcriticality, b) coolable geometry (boron concentration below solubility limit for precipitation), and c) collapsed liquid level above the top of active fuel. The TR shows that coolable geometry is retained and the collapsed liquid level remains above the active fuel height, and the Committee agrees with these conclusions.

Unlike with the US600 design, the US460 design did not request an exemption to GDC 27. Consistent with SECY-18-0099, GDC 27 has historically been interpreted as "requiring a reactor to be reliably controlled to achieve and maintain a safe, stable condition, including subcriticality, beyond the short term," and NuScale had made design changes to remain subcritical during ECCS actuation. The ability to remain subcritical after an ECCS actuation depends on the behavior of several core parameters that affect core reactivity. These include the following. An initial concentration of boron is present in the reactor coolant system at the beginning of the event and will increase (distill) in the core region due to coolant boiling during natural circulation.

Additional boron is being added from the dissolver baskets present in the containment vessel. This adds negative reactivity. The core is cooling down substantially over the 72-hour period, which adds positive reactivity. Xenon first peaks, then decays away over the 72-hour period. At 72 hours, the xenon is almost gone, which adds positive reactivity. All control rods, except the highest worth rod, are considered inserted, which adds negative reactivity. Samarium is increasing in the core over the 72-hour period. This adds negative reactivity.

It should be noted that some parameters that are considered beneficial to core cooling, such as low temperatures and low decay heat (hence low xenon), make it more difficult to remain subcritical.

The most limiting conditions to remain subcritical occur at the end of cycle, when the reactor coolant system boron concentration in the core is near zero. From the cases shown in the TR (and in Chapter 15), all analyzed cases remain subcritical, but the calculated margin to criticality can be relatively small. The smallest margin to criticality shown was 28 parts per million (ppm) boron. This calculated margin to criticality is within the predicted boron concentration uncertainty usually found in pressurized water reactors (PWRs), which is typically 50 – 100 ppm. Cold, off-normal conditions usually increase the amount of uncertainty. NuScale has indicated that there are many conservatisms built into their methodology that increase the margin to criticality, such as the use of conservative temperatures in the analysis. The NRC staff also ran computational fluid dynamic calculations that show that there is additional conservatism in the NuScale boron tracking model. In the example provided, computational fluid dynamic calculations added approximately 180 ppm to the shutdown margin. With these conservatisms, it is shown that the core remains subcritical after an ECCS actuation.

The Committee has no objection to the issuance of the SE for the topical report.

It is recommended that this writeup serves as a record of the Subcommittee meeting and that an ACRS letter report not be prepared. Further remarks will be included in the final NuScale US600 Chapter 15 Memorandum. The Committee agreed with this summary.

c. ADVANCE Act Activities Information Briefing

The Committee heard from the NRC management and staff charged with implementing the ADVANCE Act at the Agency. The Committee was updated on all activities on this important subject and discussed areas including those that could potentially impact the ACRS activities. These areas include potential changes to the staff's processes for reviewing new reactor applications and impacts on the reactor oversight process.

The Committee is committed to keeping up on these topics and as it works further to streamline its own reviews of new reactor applications.

d. Discussions During the Planning and Procedures Session

1. The Committee discussed the full committee (FC) and subcommittee (SC) schedules through September 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 March 2025. The Executive Director documented this activity in a memorandum dated April 9, 2025, ADAMS Accession No. ML25099A122.

3. The Committee briefly discussed the SC meetings that were held since the last ACRS FC meeting in February 2025, which included the following:

    - March 18 and 19: TerraPower Natrium TRs on Stability Methodology (Member Palmtag), Design Basis Transient Methodology (Member Martin), Radiological Source Term Methodology (Member-At-Large Petti), and Radiological Release Methodology (Member Martin)

    - March 20: TEUSAs Design Overview and Principal Design Criteria TR (Member Palmtag)

    - March 20: Update on Electric Power and Environmental Qualification Activities (Member Roberts)

- April 1: NuScale Chapters 1, 4, and 15 (Chairman Kirchner)

4. The Executive Director also led a discussion of draft and final RGs regarding possible review by the Committee. The Executive Director documented this activity in a memorandum dated April 9, 2025, ADAMS Accession No. ML25099A117.

5. The Executive Director also led a discussion of three topical reports that were reviewed by a lead member who gave a recommendation to the Committee about the need to review the documents. The Executive Director documented this activity in a memorandum dated April 9, 2025, ADAMS Accession No. ML25099A129.

6. Members Martin and Palmtag continued discussions of NuScale TR-0516-49416, "Non-Loss-of-Coolant-Accident Analysis Methodology," and NuScale TR-124587, "Extended Passive Cooling and Reactivity Control Methodology." The Committee agreed on the summaries that are included in item b. above.

7. Member Palmtag led a discussion about the review of TerraPower's TR on the stability methodology. On March 18, 2025, the TerraPower subcommittee of the ACRS reviewed the TerraPower, LLC TR "Stability Methodology Topical Report," NAT-9393, Revision 0. This TR describes the methodology used to evaluate the stability of the Natrium sodium-cooled fast reactor (SFR). The TR shows that power oscillations are not possible in the Natrium reactor, and this satisfies Natrium PDC 12.

   The subcommittee had the following comments:

   The FOM selected to show reactor stability is the Nyquist stability criterion. The FOM was calculated using a spatially-independent model to show that the Natrium reactor remains stable over a wide range of operating conditions. These results are expected for a tightly coupled SFR.

   The FOM was validated by applying it to the Fermi Unit 1 Oscillatory Rod Stability Measurements. The calculated results agree well with measured data, but it was necessary to perform some model adjustments. Additional validation data would give confidence that the method is robust.

   While sufficient margin was shown, the spatially-independent model is not considered a state-of-the-art methodology because it does not account for 3D power redistributions and other non-linear effects.

   It is recommended that the staff's SE be issued and that this write-up serves as the record of the subcommittee meeting and an ACRS letter report not be prepared. The Committee agreed with this summary.

8. Member Martin led a discussion of the review of the TerraPower TR on the design basis transient methodology. Member Martin reviewed TerraPower's TR NAT-9390, "Design Basis Accident Methodology for In-Vessel Events without Radiological Release," Revision 2 (ADAMS Accession No. ML24295A202), which describes the EM developed to analyze in-vessel Design Basis Accidents (DBAs) for the Natrium SFR that do not result in fuel failure or radiological release. This TR supports TerraPower's Construction Permit Application (CPA) and applies a conservative EM

framework consistent with RG 1.203 on transient and accident analysis and Nuclear Energy Institute (NEI) 18-04 for risk-informed safety analyses, as endorsed by RG 1.233.

At the March 18, 2025, ACRS Subcommittee meeting, TerraPower and NRC staff presented the EM's basis, which focuses on events, such as loss of offsite power, rod withdrawal at power, and loss of heat sink, that do not result in radiological release. The DBA methodology leverages the SAS4A/SASSYS-1 code from the U.S. DOE's Argonne National Laboratory and follows the EM Development and Assessment Process described in RG 1.203.

The development of Phenomena Identification and Ranking Tables (PIRTs) across five studies is a foundational step in guiding the focus of an EM. TerraPower's Natrium PIRTs consider fuel centerline temperature, coolant temperature, and time-at-temperature with no failure as FOMs. The PIRTs did not include state-of-knowledge rankings and PIRT participants were not formally identified, practices that are described and specified in the NRC's White Paper on the Practical Insights and Lessons Learned on Implementing Expert Elicitation (ML16287A734), which was prepared in response to Commission SRM-COMGEA-11-0001 (ML110200139). The Committee concluded that these PIRT process issues do not constitute a safety issue for the CPA. The PIRTs did recognize large SFR-specific issues like thermal stratification and shear-driven pool mixing, and TerraPower acknowledged that it is pursuing targeted testing to address existing data gaps. Physics modeling is limited to point kinetics, which puts greater burden on separate core analysis.

The NRC staff concluded that the EM addresses the first twelve EM Development and Assessment Process steps at an appropriate level of detail for a CPA, subject to limitations and conditions. They performed audits of PIRT development and the code assessment process, but did not issue RAIs or conduct confirmatory analysis.

The Committee concludes that TerraPower's methodology aligns with NRC expectations for EMs at the CPA stage and have no objection to the NRC staff's draft SE. This conclusion is supported by 1) adherence to RGs 1.203 and 1.233; 2) the EM's foundation in a mature, NRC-reviewed code; and 3) the ongoing plans to close identified gaps through testing and analysis.

It is recommended that this write-up serves as the record of the subcommittee meeting and that an ACRS letter report not be prepared.

9. Member Martin led a discussion of the TerraPower TR on the radiological release methodology. Member Martin reviewed TerraPower's TR NAT-9391, "Radiological Release Consequences Methodology," Revision 0 (ADAMS Accession No. ML24208A182), which describes the EM developed to analyze Licensing Basis Events (LBEs), DBAs, and Control Room Habitability (CRH) for the Natrium sodium fast reactor. This TR supports TerraPower's CPA and applies a risk-informed framework consistent with NEI 18-04, as endorsed by RG 1.233. It also draws from RG 1.183 for conservative DBA and CRH evaluations, and RG 1.247 for consequence metrics consistent with LBE analysis.

At the March 19, 2025, ACRS Subcommittee meeting, TerraPower and NRC staff presented the EM's technical basis. For LBE analysis, TerraPower uses the NRC-sponsored MACCS code, which includes adaptive plume segmentation and chronic dose modeling. Uncertainties are addressed through either non-parametric sampling or conservative biases. The NRC staff noted that while MACCS is well understood, its use as part of a licensing application under NEI 18-04 is first-of-a-kind.

For DBAs, TerraPower applies the RRCAT code, developed by GE Vernova and functionally similar to NRC's RADTRAD code. It uses bounding assumptions and time-averaged χ/Q dispersion factors consistent with RG 1.183. Atmospheric dispersion is not modeled dynamically in the DBA EM; instead, time-averaged, undepleted-plume χ/Q values are applied to conservatively bound site-specific variability. The same tool is used for CRH assessments, which include inhalation and submersion doses, there is also a proprietary method for gamma shine dose calculations. The control room is designated as non-safety-related with special treatment, with conservative assumptions ensuring compliance with TerraPower's Principal Design Criteria 19 (ADAMS Accession No. ML24101A362).

An appendix to the report presents a method for Emergency Planning Zone sizing based on the LBE framework. While not a required EM feature, it provides useful insight into TerraPower's plan for a risk-informed basis for assessing plume exposure pathway emergency planning boundaries.

The NRC staff determined that NAT-9391, Revision 0, describing their DBA, LBE, and CRH radiological release EMs, provides an acceptable approach for analyzing site-specific radiological release consequences, subject to two limitations and conditions. The NRC staff also noted that they did not evaluate the specific acceptability of TerraPower's proprietary RRCAT code but confirmed its consistency with regulatory guidance. Finally, the NRC emphasized that approval of this TR does not extend to the acceptability of the methodology for determining the plume exposure pathway emergency planning zone and source terms, which are addressed in separate submittals.

The Committee finds the EMs described in TerraPower's TR NAT-9391 to be consistent with NRC expectations for risk-informed radiological consequence analysis at the CPA stage and has no objection to the NRC staff's draft SE. This conclusion is supported by their use of established NRC-endorsed tools for LBE evaluation and the application of RG 1.183 principles for DBA and CRH scenarios.

It is recommended that this write-up serves as the record of the subcommittee meeting and that an ACRS letter report not be prepared.

10. Member Bier led a discussion about a subcommittee meeting on artificial intelligence planned for November 2025. The Human Factors Reliability & Probabilistic Risk Assessment (PRA), and Digital Instrumentation and Control and Electrical Subcommittees are arranging the meeting and have reached out to Professor David Woods from Ohio State as a possible speaker; other individuals may be contacted if Dr. Woods is unavailable.

11. Vice Chairman Halnon led a discussion of the planned visits to the Seabrook Nuclear Station to further investigate the Alkali Silica Reaction (ASR) issue and to the Westinghouse Newington site, April 17 and 18, 2025. There will be a Plant Operations Subcommittee meeting about the Seabrook ASR topic on April 17, 2025.

    Logistics were discussed as well as the need to complete security forms to access the Seabrook Nuclear Station.

12. Member Roberts led a discussion of the review plan for the Kemmerer CPA, which is for a TerraPower Natrium reactor.

    The CPA for the Kemmerer site was submitted in March 2024. The NRC staff have requested we issue our final letter by April 2026. This will require all subcommittee reviews to be done by early March 2026. The staff is working on their proposed schedule to present the CPA to the Committee, but notionally the staff may be ready for the first meeting in August 2025, with the remaining subcommittee meetings from October 2025 through early March 2026.

    It should be noted that the Preliminary Safety Analysis Report uses the Technology-Inclusive Content of Application Project/Advanced Reactor Content of Application Project organization method, and our previous chapter review process will need to be altered. The staff proposed an approach using seven SC meetings. Of note, the approach covers Chapter 3 (Licensing Basis Events) in three different meetings, due to it being the foundation of the Licensing Modernization Program approach, and it requires two meetings to cover Chapter 7 (Descriptions for safety-significant structures, systems and components (SSCs)) due to its wide scope (roughly equivalent to Chapters 4-10 in a traditional Safety Analysis Report). As the Committee gets closer to its reviews, other Chapters may also be covered in multiple meetings; for example, Chapter 5 (Safety Functions, Design Criteria, and SSC safety classification) may need to be covered along with the final Chapter 3 details if such details would affect safety classification of SSCs. The Committee will review and refine the process throughout the review as experience is gained.

    Member Roberts suggested the following adjustments to the ACRS review process:

    - After each meeting that covers a chapter, the lead member will determine if there is enough information to write a chapter memo, if so, a chapter memo will be written. If there is not enough information to write a chapter memo, a meeting summary will be written for the planning and procedures meeting that describes any issues that need to be addressed by the applicant/staff in future meetings.

    - For Chapters 3, 7, and 11, the lead members (Member Martin, Member Harrington, and Vice Chairman Halnon respectively) will determine the best way to include all relevant input in their memo. This may involve memo attachments to be referenced in a cover memo, combination of inputs from others into a single memo, or whatever approach is most efficient for that chapter.

13. Chairman Kirchner led a discussion about review of the NuScale standard design approval application chapter memoranda for Chapter 6, “Engineered Safety

Features" (Member Harrington) and Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation" (Member Dimitrijevic). The memos were finalized.

14. Member Bier led a discussion about an upcoming subcommittee meeting on the use of PRA on advanced reactor applications. Members Bier and Dimitrijevic will provide information to the Committee about plans to discuss various issues with the NRC staff at a SC meeting on May 21, 2025. During the ACRS June 2024, Planning & Procedures Subcommittee meeting, Vice Chairman Halnon, Member Bier, and Member Dimitrijevic brought up the topic: How PRA is being used with reactors that have a very low risk profile (NuScale, SMR-300, etc.). This came up during the discussion of a HOLTEC SMR-300 topical report on Risk Significance Determination Methodology. During the discussion, ACRS Members agreed that there may be larger policy/technical issues that should be considered outside of the Holtec TR. The ACRS PRA and Regulatory Policy SC had two informal meetings with the staff from the Office of Nuclear Reactor Regulation and Office of Nuclear Regulatory Research to discuss whether a SC should be held to address technical issues, and what should be covered. Examples include the use of risk importance measures for advanced reactors, whether there is a consistent approach (or recommended approach) for new applications, and whether that approach could permit situations in which a plant's low risk profile (low core damage frequency/large release frequency) is not adequately assured.

    Staff agreed to a SC meeting and/or a series of SC meetings on the use of PRA analysis for new/advanced reactor reviews. The first meeting is open to the public and is scheduled for May 21, 2025. After that, it will be determined whether an additional SC meeting is needed, and whether there is any recommendation for a letter.

    At the May 2025 SC meeting, Members Bier and Dimitrijevic will open the SC meeting with a discussion of the objective and scope. Both Members will present a brief slide presentation of the technical issues that have been identified as needing to be addressed for advanced reactor applicants. Staff will present on:

    - Relative and Absolute Risk Metrics,
    - PRA Completeness,
    - Staff and Industry Guidance Under Development,
    - Cliff Edge Effects, and
    - Uncertainty Analysis.

    Additionally, there will be a presentation by ACRS Consultant Bley on cliff edge effects. Future SC meetings could possibly involve stakeholders. Additionally, the SC will determine whether to recommend that the FC should write a Letter Report after all SC meetings have been held.

15. ACRS staff member Burkhart led a discussion about the information exchange on the topic of sodium fast reactors scheduled for the August 2025 subcommittee week. At this activity, subject matter experts from both Argonne National Laboratory and Idaho National Laboratory will share relevant information with the Committee.

16. Member Roberts led a discussion of the update on the reconciliation for the TerraPower Natrium topical report on Emergency Planning Zone sizing methodology. The evaluation of the NRC staff's response is ongoing, and the reconciliation will likely be completed for the May or June 2025 full committee meeting.

17. Under additional topics, it was discussed that the visit to the BWXT fuel fabrication facility may not be possible. It was noted that this week could be taken by the TerraPower Natrium Kemmerer CPA review activities instead.

18. A closed session was conducted to discuss proprietary and administrative information.

19. The following topics are on the agenda of the 725th ACRS Full Committee meeting, which will be held on May 7 through 9, 2025:

    - NuScale SDAA topics including the final letter report on the application,

    - TerraPower Natrium topical report on the radiological source term methodology, and

    - Seabrook Station alkali-silica reaction topic.

Sincerely,

Walter L Kirchner Signed by Kirchner, Walter on 05/16/25

Walter L. Kirchner
Chairman

Enclosure:
List of Acronyms

May 16, 2025

SUBJECT: SUMMARY REPORT – 724th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, APRIL 2-3, 2025

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LIST OF ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
ANS	American Nuclear Society
ANSI	American Nuclear Standards Institute
ASR	Alkali Silica Reaction
CPA	Construction Permit Application
CRH	Control Room Habitability
DBAs	Design Basis Accidents
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
EM	Evaluation Method
FC	Full Committee
FOM	Figures-of-Merit
GDC	General Design Criteria
IMSR	Integrated Molten Salt Reactor
L&C	Limitations and Conditions
LBEs	Licensing Basis Events
LOCA	Loss-of-Coolant Accident
MSR	Molten Salt Reactor
MWth	Megawatt Thermal
NEI	Nuclear Energy Institute
Non-LOCA	Non-Loss of Coolant Accident
NRC	Nuclear Regulatory Commission
NPM	NuScale Power Module
PDC	Principal Design Criteria
PIRT	Phenomena Identification and Ranking Tables
PPM	Parts per Million
PRA	Probabilistic Risk Assessment
RG	Regulatory Guide
SC	Subcommittee
SE	Safety Evaluation
SDAA	Standard Design Approval Application
SFR	Sodium-Cooled Fast Reactor
SSC	Systems, Structures and Components
TEUSA	Terrestrial Energy USA, Inc.
TR	Topical Report
XPC	Extended Passive Cooling