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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 28, 2026

Mr. Charles V. McFeaters
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 – SUMMARY
OF REGULATORY AUDIT IN SUPPORT OF LICENSE AMENDMENT
REQUEST REGARDING COBALT-60 PRODUCTION (EPID L-2025-LLA-0154)

Dear Mr. McFeaters:

By letter LR-N25-0074 dated September 24, 2025 (Agencywide Document Access and Management System (ADAMS) Accession No. ML25268A072), PSEG Nuclear LLC (the licensee) requested to amend Renewed Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2, respectively. The licensee requested to revise the operating licenses to allow production of cobalt-60.

The U.S. Nuclear Regulatory Commission (NRC) staff conducted a virtual audit to support its review of the license amendment request (LAR). The NRC staff issued its audit plan on February 3, 2026 (ML26029A125). The NRC staff conducted the audit using virtual meetings and an internet-based-portal provided by the licensee and its contractor, Westinghouse Electric Company, LLC (WEC). Using that portal, the NRC staff reviewed documents (e.g., calculations and reports) related to the LAR but not available in ADAMS on the Salem dockets. During the audit, the staff met virtually with the licensee mostly on a weekly basis. The staff used these meetings to confirm its understanding of the LAR, discuss the documents in the portal, and decide whether any information needed to be submitted on the docket to complete the NRC staff's safety evaluation.

As discussed in the NRC staff's audit plan supplement dated June 1, 2026 (ML26139A176), the NRC staff also conducted a portion of the audit at WEC's office in Cranberry Township, Pennsylvania, during the week of June 8, 2026. The in-person audit focused on the computational fluid dynamics methodologies, calculations, and validation.

**Enclosures 2 and 4 to this summary contain proprietary information.
When separated from Enclosures 2 and 4, this document is DECONTROLLED.**

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During the audit, the NRC staff identified information needed on the docket to support its review and verbally communicated these needs to the licensee during the audit calls. The licensee voluntarily submitted supplements dated June 24, 2026 (ML26175A183), and August 31, 2026 (ML26243A375) to address these information needs. The NRC's licensing project manager informed licensee staff by email on September 18, 2026, that the NRC staff had completed its audit. There were no open items resulting from the audit.

Enclosure 1 to this audit summary lists the individuals that participated in the audit. Enclosure 2 (proprietary) and Enclosure 3 (nonproprietary) list the NRC staff's observations and supplemental audit requests. Enclosure 4 (proprietary) and Enclosure 5 (nonproprietary) list the documents audited by NRC staff.

If you have any questions, please contact me at (301) 415-0489 or by e-mail at Audrey.Klett@nrc.gov.

Sincerely,

/RA/

Audrey Klett, Senior Project Manager
Operating Reactor Licensing and Data Branch 1
Division of Licensing Projects I
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:

1. List of Audit Participants
2. Proprietary Observations and Supplemental Audit Requests
3. Nonproprietary List of Supplemental Audit Requests
4. Proprietary List of Documents Audited
5. Nonproprietary List of Documents Audited

cc: without Enclosures 2 and 4: Listserv

C. McFeaters

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SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 –SUMMARY
OF REGULATORY AUDIT IN SUPPORT OF LICENSE AMENDMENT
REQUEST REGARDING COBALT-60 PRODUCTION (EPID L-2025-LLA-0154)

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ADAMS Accession Nos.:

ML26264A098 (Package)

ML26271A070 (Non-Proprietary)

20260921-02517 (e-Concurrence Case)

LIST OF AUDIT PARTICIPANTS

U.S. Nuclear Regulatory Commission Staff

Audrey Klett
Ahsan Sallman
Clint Ashley
David Brown
David Coy
David Garon-Candelaria
Ernesto Quinones
George Wang
Jack Vande Polder
John Lehning
Noushin Amini
Raul Hernandez
Richard Fu
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Mark Smith
Michael Conner
Tatiana Siaraferas
Virginia Lynch

OBSERVATIONS AND SUPPLEMENTAL AUDIT REQUESTS

While PSEG Nuclear LLC (the licensee) subsequently submitted docketed supplemental information discussed during the audit that the U.S. Nuclear Regulatory Commission (NRC) staff found necessary to support regulatory conclusions, several additional detailed observations the NRC staff deemed relevant to the review are documented below.

- The licensee's submittal referred to the [[

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- The computational fluid dynamics (CFD) simulations performed by the licensee used [[

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- As indicated in Figure 6 of Westinghouse calculation note CN-NFPE-23-004, Revision 3, the licensee used a [[

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The U.S. Nuclear Regulatory Commission (NRC) staff's initial request of information to audit (i.e., Audit Item Nos. 1 through 21) was provided in its audit plan on February 3, 2026 (Agencywide Document Access and Management System Accession No. ML26029A125). The staff also performed an in-person audit at Westinghouse Electric Company, LLC facilities in June 2026, as described in the supplemental audit plan dated June 1, 2026 (ML26139A176). The following items reflect new requests and requests related to the previously requested items that were discussed with the licensee during the audit.

ITEM NO.	AUDIT REQUEST
1	NRC Reviewer: Clinton Ashley The NRC staff requested to audit the revised technical specifications (TS) page.
2	NRC Reviewers: Noushin Amini and John Lehning The NRC staff requested various clarifying information and descriptions of the computational fluid dynamics (CFD) methodology, calculation inputs, and considerations; final CFD calculations, margin calculations, and a summary of the updated [[]]. The staff requested discussions on the CFD calculation process, including: key supporting inputs; [[]], and ways of addressing concerns with validation of calculations. The staff requested if [[]], and what assumptions are being applied relative to the CFD calculation methodology. The staff requested to audit a nodalization diagram and receive a discussion of the [[]].
3	NRC Reviewer: Ernesto Quinones The NRC staff requested to audit the revised license conditions.
4	NRC Reviewer: Ahsan Sallman The NRC staff requested the net positive suction head (NPSH) analysis for the loss-of-coolant accident (LOCA) recirculation phase. The staff requested additional confirmatory information, as discussed in the licensee's supplement dated June 24, 2026 (ML26175A184).
5	NRC Reviewer: Ahsan Sallman The NRC staff requested the calculation for the development of [[]] and a summary of how this decay heat was applied in the containment analysis. The staff requested additional confirmatory information, as discussed in the licensee's supplement dated June 24, 2026.

ITEM NO.	AUDIT REQUEST
6	NRC Reviewers: John Lehning and Thomas Scarbrough The staff requested a summary document on how COBA decay heat is treated in each of the safety analyses described in the license amendment request (LAR). The NRC staff further requested specific discussion concerning whether the cobalt burnable absorbers (COBAs) have an effect on peak cladding temperature (PCT) and any local impacts and neighboring rod assumptions in the Appendix K LOCA evaluation models.
8	NRC Reviewer: Richard Fu The NRC staff requested the calculation for the creep analysis.
11	NRC Reviewer: John Lehning and Richard Fu The NRC staff requested more information on capsule quality and operating history.
12	NRC Reviewer: John Lehning The NRC staff requested the methods used for calculations to determine the response of COBAs to a postulated rod ejection accident and clarification on whether identified methods are approved or were modified. The NRC staff requested clarification on [[]]
13	NRC Reviewer: John Lehning The NRC staff requested clarification on whether the [[]] The NRC staff also requested explanation of details of the licensee's supporting calculations, including (1) the difference between Case 2 and the other case in reference 3 to CN-NFPE-21-009 and (2) why Case 2 is most realistic or appropriate to use.
15	NRC Reviewer: Summer Sun The NRC staff requested numerical comparisons of the heat flux hot channel factor (FQ), the nuclear enthalpy rise hot channel factor (FΔH), axial offset (AO), and axial shape parameters for “with-COBA” and “without-COBA” cases, demonstrating that the differences are minimal and within typical designer variability. The NRC staff requested discussion on specific sensitivity studies performed, specifying what parameters were varied and referencing plots or tables from CN-CPBA-19 showing that axial shape differences are negligible. The NRC staff requested discussion on acceptance criteria or thresholds used to judge that the COBA’s impact is minimal and clarification on whether if any localized increases in axial or radial peaking were observed in COBA-bearing assemblies across burnup. The NRC staff requested discussion on whether there are any cycle-specific conditions or core designs where the COBA’s impact could be non-minimal, including the specific reload safety analysis checklist (RSAC) and TS checks run on COBA-included patterns, demonstrating compliance with WCAP-9272 and ensuring no hidden safety parameter deviations.

ITEM NO.	AUDIT REQUEST
	Additional audit requests included the following: A. Quantitative Results and Sensitivity Analyses 1. Provide quantitative comparisons of FQ, FΔH, AO, and key axial power shape metrics between the with-COBA and without-COBA cases for each exposure point evaluated. Include tabulated and graphical results or reference to the applicable calculations. 2. Identify any limiting cases and explain whether COBA changes margin significantly. 3. Provide sensitivity studies demonstrating the impact of: (1) COBA enrichment variations; (2) COBA axial placement assumptions; and (3) COBA burnup modeling assumptions, on core peaking parameters and axial power shape. B. Methodology and Model Justification 1. Describe and justify the modeling methodology used to ensure that COBA-induced flux depressions and spectral shifts are correctly captured in nodal diffusion calculations, cross-section interpolation schemes, and any physics models. 2. Identify all NRC-approved methodologies used in the calculations, and justify their applicability to COBA-containing cores. C. Axial Power Shape and Operational Limits 1. Provide detailed axial power distributions (shape plots and normalized tables) from the COBA and non-COBA cases, quantify deltas at each axial node, and identify how COBA affects top-peaking or bottom-peaking behavior. 2. Explain how axial offset limits were confirmed for COBA cycles. Provide AO operating bands, and discuss cycle-specific AO uncertainty or variation due to COBAs. 3. Discuss supporting calculations demonstrating whether COBA impacts shutdown margin evaluations, rod insertion limits, and anticipated operational occurrences (AOOs) sensitive to axial peaking (e.g., control element assembly withdrawal events or dropped rod events). These analyses are referenced in the non-LOCA transient review framework in section 8.3 of attachment 2 to the license amendment request (LAR). D. Validation, Fuel Management, and Design Margin 1. Provide justification that the “minimal differences” claimed between COBA and non-COBA cores remain within fuel design limits, reload safety analysis acceptance criteria, and statistical and modeling uncertainty margins. 2. Describe the fuel-management adjustments applied to maintain similarity to the original loading pattern, and provide specific enrichment changes and assembly shuffle details. Address whether alternative loading strategies could produce more adverse COBA effects. 3. Provide cycle-specific safety confirmation checks (referenced qualitatively in the response) summarizing limiting departure from nucleate boiling ratio (DNBR) locations for each case, limiting linear heat generation rate (LHGR) or maximum average planar LHGR (MAPLHGR) parameters, and any COBA-unique peaking behavior.

ITEM NO.	AUDIT REQUEST
16	NRC Reviewer: John Lehning The NRC staff requested that WCAP-8611 be added to the audit portal and for the response to include information from the calculation that is responsive to the initial audit request concerning the potential for [[]] as relevant to Salem's plant-specific conditions during a postulated LOCA.
20	NRC Reviewer: Summer Sun Provide a reference to the NRC approval of the thermal-hydraulic (T-H) method, and address applicability of the method to the core with COBA assemblies installed. Provide rationale to support that the [[]] Provide reference of the NRC approval of the T-H subchannel analysis methods or the topical report documenting the above-mentioned DNBR correlation, T-H subchannel codes, and statistical methods method. Address applicability of the method to the core with COBA assemblies installed.
22	NRC Reviewer: David Brown The LAR indicates that [[]] Can the licensee explain why [[]] is an acceptable value, in terms of the anticipated [[]]
23	NRC Reviewer: David Brown How do the procedures for offering the package for transport to a carrier address the certificate requirements for adequate heat dissipation?
24	NRC Reviewer: Thomas Scarbrough How has the potential impact of the heat load generated by the COBAs on the cooling water flow requirements during normal, abnormal, and accident conditions of the reactor coolant system and spent fuel pool been addressed to support the LAR?

ITEM NO.	AUDIT REQUEST
25	NRC Reviewer: Summer Sun The NRC staff requests the licensee to discuss the cycle-specific methodology for confirming that the American Nuclear Society (ANS) standard ANS 5.1–1979 +2σ decay heat remains conservative with COBAs present, including treatment of activated isotopes, uncertainty quantification, and acceptance criteria. 1. Treatment of Activated Isotopes in COBA Assemblies. Provide a detailed description of how activated isotopes associated with COBA inserts are modeled in the decay-heat calculations, including assumptions, data sources, activation pathways, and any benchmarking performed. 2. Demonstration of Conservatism Relative to ANS-5.1-1979 + 2σ. Provide the cycle-specific approach used to confirm that total decay heat (including COBA inserts) remains below or bounded by ANS-5.1-1979 + 2σ. Include methodology for comparison, acceptance criteria and margin definition, and representative sample calculations or bounding assessments. 3. Uncertainty Quantification in the Presence of COBA Assemblies. Clarify how the revised uncertainty methodology accounts for decay-heat modeling of COBA assemblies. Identify whether the uncertainty value applies uniformly or if COBA-specific uncertainty is evaluated separately. 4. Acceptance Criteria for Decay Heat Beyond SFP Thermal Limits. The response cites SFP temperature limits (149 degrees Fahrenheit (°F)/180 °F) but does not identify decay-heat acceptance criteria linked to methodology validation. Provide the acceptance criteria used to determine whether cycle-specific decay-heat predictions, including COBA effects, are considered valid, conservative, and acceptable for use in safety analyses. 5. Verification of Safety Analysis Margins Affected by COBA Decay Heat. The response describes procedural controls (CC-AA-102, NF-AA-100) but does not identify the [[]] Provide a list of safety analyses whose margins could be influenced by COBA decay heat and describe how each analysis is evaluated during reload and configuration-change reviews to ensure continued margin adequacy. 6. [[]] 7. Is the response applicable to non-LOCA analyses?

ITEM NO.	AUDIT REQUEST
26	NRC Reviewer: Geroqe Wang Regarding spent fuel pool (SFP) configuration and fuel assembly location/harvesting workstation information, the NRC staff requests the licensee to provide drawings or figures showing the relative configuration of fuel assemblies containing COBAs within the SFP, including: (a) SFP floor and wall thicknesses and corresponding elevations, and (b) the layout of the fuel assembly/harvesting workstation within the SFP, their top and bottom elevations, and the horizontal distance from the fuel assembly to the SFP wall. (a) Provide the total incident energy flux and the total integrated radiation dose at the interior face of the SFP walls and floor for both (1) storage of COBA fuel assemblies and (2) harvesting workstation structure. (b) Table 9.3-1 of LR-N25-0074, Attachment 3, identifies that the incidental gamma flux exceeds the NUREG/CR-6927 criteria for Cases 1 and 8. In the response to AI26 under the “Gamma Heating” discussion for the Cast Pit, the licensee indicated that even when the incident gamma flux exceeded the acceptance criterion, the resulting SFP concrete temperatures remained within the limits set by Section A.4.1 of ACI 349-01. For Cases 1 and 8, provide the technical justification demonstrating how the SFP concrete temperatures remain within ACI 349-01 limits despite these exceedances.
27	NRC Reviewer: Geroqe Wang Regarding gamma flux and dose calculation package, the NRC staff requests the licensee to provide the calculation package used to determine gamma flux and dose within the SFP, including at the fuel assembly location and harvesting workstation, and on the faces of SFP walls and floor. Include the methodology (assumptions, input parameters) and technical basis used for gamma flux and dose estimation.
28	NRC Reviewer: Geroqe Wang Regarding SFP structural analysis/evaluation for additional loading and temperature above 150 °F, the NRC staff requests the licensee to provide the LAR-related SFP structural calculations supporting any additional loading conditions, including any applicable load-drop cases, and SFP concrete temperature above 150 °F.
29	NRC Reviewer: Geroqe Wang Regarding temporary storage stand documentation, the NRC staff requests the licensee to provide design drawings and structural calculations for the temporary storage stands and harvesting workstation within the SFP.
30	NRC Reviewer: Geroqe Wang Regarding as-built construction documentation, the NRC staff requests the licensee to provide as-built construction drawings for the SFP concrete structure and stainless steel liner, including all relevant plans, sections, elevations, and construction details.

ITEM NO.	AUDIT REQUEST
31	NRC Reviewer: Geroge Wang Regarding concrete core sample testing and reports, the NRC staff requests the licensee to provide the reports and testing results for the two concrete core samples taken from the Salem, Unit 1 SFP in 2013 and 2015 (refer to Unit 1 TS, page 9).
32	NRC Reviewer: Geroge Wang Regarding corrective actions and work documentation, the NRC staff requests the licensee to provide condition reports, work orders, and photographs associated with any significant issues with the SFP concrete structure and liner (including any inspections, repairs, or modifications) in the past ten years.
33	NRC Reviewer: Geroge Wang Section 3.8.4.4, item 7 of the updated final safety analysis report (UFSAR, Revision 35, pdf page 478) discusses evaluations for the spent fuel pool structure (reinforced concrete as well as the stainless steel liner and its anchors). Provide a discussion, including quantitative results, demonstrating how the structural acceptance criteria were met for the SFP concrete structure and the stainless steel liner for the controlling load combinations, including any impacts of increased thermal loads and additional loads introduced by the storage of COBA fuel assemblies and the harvesting workstation structure. The NRC staff requests the response to include a concluding statement that explicitly affirms that all relevant loads remain within the current design basis. The staff requests confirmation of whether all loads associated with COBA implementation and the harvesting workstation, including seismic and thermal loads, are fully bounded by the current design-basis loads evaluated in UFSAR, section 3.8.4.4, item 7 for the spent fuel pool concrete structure, stainless-steel liner, and liner anchorage system.
34	NRC Reviewer: Geroge Wang Provide the associated calculations supporting the analysis of the SFP concrete structure and the stainless steel liner. Also provide the as-built SFP plan view and cross sections drawings in both directions to support the staff's review of the SFP calculations.
35	NRC Reviewer: Ahsan Sallman "Component Sizing Evaluation," of attachment 2 to the LAR states, in part, "The insertion of cobalt-59 target material, which is used to produced cobalt-60, in fuel assembly thimbles does not impact the control systems operability since the control systems logic, setpoints and time constants, and the NSSS [nuclear steam supply system] capacities and performance capabilities are not being revised." Identify the specific control systems whose logic, setpoints and time constants, and NSSS capacities and performance capabilities were assessed to conclude that control system operability would not be impacted. The staff also asked for additional confirmatory information, as discussed in the licensee's supplement dated August 31, 2026.

ITEM NO.	AUDIT REQUEST
36	NRC Reviewer: Ahsan Sallman The second paragraph in section 8.4 of attachment 3 to the LAR states, in part, “The current Loss of RHR [residual heat removal] at Midloop analysis has sufficient margin to account for the additional decay heat from COBAs; therefore, there is no impact to the Loss of RHR at Midloop.” The third paragraph in that section states, in part, “The current analysis has sufficient margin to account for the additional decay heat from COBAs; therefore, there is no impact to NCC [natural circulation cooldown].” Provide the current NCC analysis showing margins and evaluation of margins in the presence of COBA decay heat. The staff also asked for additional confirmatory information, as discussed in the licensee’s supplement dated August 31, 2026.
37	NRC Reviewer: Ahsan Sallman The first paragraph in section 8.5 of attachment 3 to the LAR states, in part, “A generic impact evaluation was performed to determine the conservatism of various decay heat models relative to the addition of COBA decay heat. The result of the generic impact evaluation was a set of screening criteria to determine that the results of the related analyses bound the use of COBAs. The screening criteria are as follows:” Provide details of the generic impact evaluation from which the screening criteria 1 through 4 in section 8.5 are developed.
38	NRC Reviewer: Ahsan Sallman The second paragraph in section 8.5 of attachment 3 to the LAR states, “FLEX [Flexible Coping Strategies] analyses used the ANS -5.1-1979 σ [sigma] = 2 model for all FLEX analyses; therefore, there is no impact to the FLEX analyses due to the addition of COBA.” Screening criterion 2 of section 8.4 states, “Any analyses for 72 hours or less using ANS-5.1-1979 σ = 1 or σ = 2 or BOP-FR-8 are considered conservative and do not require re-analysis.” (a) List all FLEX analyses that use the ANS-5.1-1979 σ = 2 decay heat model. (b) Which analysis of record (AOR) uses the less conservative BOP-FR-8 decay heat model compared to the ANS-5.1-1979 σ = 2 model? Provide reasons for using the less conservative decay heat model. The staff also asked for additional confirmatory information, as discussed in the licensee’s supplement dated August 31, 2026.

LIST OF DOCUMENTS AUDITED

The NRC staff reviewed, in a sampling manner, the following licensee documents during its audit (this list does not include documents available in ADAMS):

AI01 (Audit Item 01)

1. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 1," April 27, 2026.
2. PSEG Nuclear LLC, technical specification markup, "TS mark up for Design Features for COBA," May 1, 2026.

AI02

3. Westinghouse Electric Company LLC, presentation slides, "NRC Meeting 05072026," May 7, 2026.
4. Westinghouse Electric Company LLC, presentation slides, "May 28 Co-60 Audit Presentation Slides," May 28, 2026.
5. Westinghouse Electric Company LLC, presentation slides, "NRC Audit Wrapup 2026-06-11 CFD Activities Final," June 11, 2026.
6. Westinghouse Electric Company LLC, audit response memorandum, "Audit Item 2 In-Person Audit Summary, Revision 6," July 8, 2026.
7. Westinghouse Electric Company LLC, presentation slides, "NRC Meeting Slides 6-30-2026," June 30, 2026.
8. Westinghouse Electric Company LLC, presentation slides, "2026-06-30 CFD Activities Summary," June 30, 2026.
9. Westinghouse Electric Company LLC, presentation slides, "Audit Item 2 Summary of June30 CFD Presentation," June 30, 2026.
10. Westinghouse Electric Company LLC, audit response memorandum, "Overview – Audit Item 2 – Consolidated Response Outline," August 17, 2026.
11. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 2, Revision 7," August 21, 2026.
12. Westinghouse Electric Company LLC, audit response memorandum, "AI02 Overview CFD summary 08212026," August 21, 2026.
13. Westinghouse Electric Company LLC, audit response memorandum, "AI02 Overview CFD summary 08252026," August 25, 2026.
14. Westinghouse Electric Company LLC, presentation slides, "CFD Status for NRC Audit 2026-04-20," April 20, 2026.
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AI03

51. PSEG Nuclear LLC, license or technical specification markup, "AI03 OL Markups Revised," April 6, 2026.
52. PSEG Nuclear LLC, license or technical specification markup, "AI03 OL Markups Revised-Clean Version," May 4, 2026.
53. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 3, Revision 2," May 4, 2026.

AI04

- 54. Westinghouse Electric Company LLC, WCAP-18659, Revision 0, technical report, "Reactor Internals Upflow Conversion Program Engineering Report for Salem Nuclear Generating Station Unit 2," August 26, 2021.
- 55. Westinghouse Electric Company LLC, WCAP-18707, Revision 0, technical report, "Reactor Internals Upflow Conversion Program Engineering Report for Salem Nuclear Generating Station Unit 1," May 31, 2022.
- 56. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 4," April 23, 2026.

AI05

- 57. PSEG Nuclear LLC, S-C-RHR-MDC-1711, Revision 7, design analysis, "Available NPSH at RHR Pump in Recirculation Mode," July 8, 2022.
- 58. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 5, Revision 1," April 8, 2026.
- 59. Westinghouse Electric Company LLC, audit response memorandum, "Sallman - AI04-05 Response Confirmations, Revision 1," May 11, 2026.
- 60. [[

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AI06

- 62. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 6, Revision 2," July 7, 2026.
- 63. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 6, Revision 3," July 31, 2026.
- 64. [[

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AI07

- 78. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 7," April 23, 2026.
- 79. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 7, Revision 1," June 25, 2026.
- 80. [[

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AI08

85. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 8, Revision 1," April 23, 2026.

86. [[

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AI09

87. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 9," February 17, 2026.

88. [[

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AI10

91. Westinghouse Electric Company LLC, Revision 1, audit response memorandum, "Overview -- Audit Item 10, Revision 1," April 2, 2026.

92. [[

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AI11

94. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 11, Revision 1," April 2, 2026.

95. [[

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AI12

96. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 12, Revision 2," May 12, 2026.

97. [[

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AI13

- 101. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 13, Revision 1," April 23, 2026.
- 102. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 13, Revision 2," June 24, 2026.
- 103. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 13, Revision 3," July 2, 2026.
- 104. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 13, Revision 4," to July 10, 2026.
- 105. [[

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AI14

- 107. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 14, Revision 1," May 12, 2026.
- 108. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 14, Revision 2," July 16, 2026.
- 109. Westinghouse Electric Company LLC, audit response memorandum, "Additional Information -- Audit Item 14," June 10, 2026.
- 110. PSEG Nuclear LLC, NF-AA-100, Revision 18, procedure, "Reload Control Procedure."
- 111. PSEG Nuclear LLC, NF-AP-100-7000, Revision 19, procedure, "Westinghouse NSSS Reload Design Control Implementation."

AI15

- 112. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 15, Revision 2," May 15, 2026.
- 113. [[

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AI16

- 115. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 16, Revision 1," May 11, 2026.
- 116. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 16, Revision 2," June 10, 2026.
- 117. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 16, Revision 3," June 18, 2026.
- 118. [[

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AI17

- 122. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 17," April 23, 2026.
- 123. [[

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AI18

- 125. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 18," February 17, 2026.

AI19

- 126. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 19, Revision 2," May 11, 2026.
- 127. [[

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AI20

- 130. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 20, Revision 2," April 22, 2026.
- 131. Westinghouse Electric Company LLC, WCAP-15025, technical report, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.
- 132. Westinghouse Electric Company LLC, WCAP-14565, technical report, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.
- 133. Westinghouse Electric Company LLC, WCAP-18240, Revision 0, technical report, "Westinghouse Thermal Design Procedure (WTDP)," April 2020.
- 134. Westinghouse Electric Company LLC, WCAP-9272, technical report, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
- 135. [[

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AI21

- 136. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 21, Revision 1," April 6, 2026.
- 137. [[

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AI22

- 138. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 22," March 31, 2026.

AI23

- 139. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 23," March 24, 2026.

AI24

- 140. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 24," April 8, 2026.

AI25

- 141. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 25, Revision 4," July 7, 2026.
- 142. Westinghouse Electric Company LLC, audit response memorandum, "Overview -- Audit Item 25, Revision 5," August 3, 2026.
- 143. Westinghouse Electric Company LLC, audit response memorandum, "Audit Item 25 Response for LOCA PCT Impact," July 23, 2026.
- 144. PSEG Nuclear LLC, OU-SA-105, procedure, "Shutdown Safety Management Program – Salem Annex."
- 145. PSEG Nuclear LLC, S-C-SF-MDC-1800, Revision 17, design analysis, "Decay Heat-up Rates and Curves," September 19, 2024.
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