



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

September 29, 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 – ISSUANCE
OF AMENDMENT NOS. 357 AND 339 REGARDING COBALT-60 PRODUCTION
(EPID L-2025-LLA-0154)

Dear Mr. McFeaters:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 357 and 339 to Renewed Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the operating licenses and technical specifications in response to PSEG Nuclear LLC's application dated September 24, 2025, as supplemented by letters dated June 24, 2026, and August 31, 2026.

These amendments revise the renewed facility operating licenses and the technical specifications to allow irradiation of cobalt-59 and the production and transfer of cobalt-60.

**Enclosure 3 to this letter contains proprietary information.
When separated from Enclosure 3, this document is DECONTROLLED.**

- 2 -

A copy of the related safety evaluation is also enclosed. A notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

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. Amendment No. 357 to
License No. DPR-70
2. Amendment No. 339 to
License No. DPR-75
3. Proprietary Safety Evaluation
4. Nonproprietary Safety Evaluation

cc: Listserv without Enclosure 3



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

PSEG NUCLEAR LLC

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 357
Renewed License No. DPR-70

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Constellation Energy Generation, LLC (the licensees) dated September 24, 2025, as supplemented by letters dated June 24, 2026, and August 31, 2026, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to license conditions and technical specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 357, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications, and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Wrona, Chief
Operating Reactor Licensing and Data Branch 1
Division of Licensing Projects I
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed
Facility Operating License
and Technical Specifications

Date of Issuance: September 29, 2026

ATTACHMENT TO LICENSE AMENDMENT NO. 357

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following pages of Renewed Facility Operating License No. DPR-70 with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove

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Insert

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Replace the following page of the Appendix A, Technical Specifications, with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove

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Insert

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- H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Renewed Facility Operating License No. DPR-70, subject to the conditions for protection of the environment set forth in the Technical Specifications, Appendix B is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied;
 - I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30,40, and 70 including 10 CFR Sections 30.33, 40.32, and 70.23 and 70.31;
 - J. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21 (c), such that there is reasonable assurance that the activities authorized by this renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations; and
 - K. The production, possession, and transfer of Cobalt-60 as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30. This type of radioactive material is considered to be bulk material (ML25100A186).
2. Renewed Facility Operating License No. DPR-70, is hereby issued to PSEG Nuclear LLC, and Constellation Energy Generation, LLC (the licensees), to read as follows:
- A. This renewed license applies to the Salem Nuclear Generating Station, Unit No. 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by PSEG Nuclear LLC and Constellation Energy Generation, LLC and operated by PSEG Nuclear LLC. The facility is located on the applicants' site in Salem County, New Jersey, on the southern end of Artificial Island on the east bank of the Delaware River in Lower Alloways Creek Township, and is described in the "Final Safety Analysis Report" as supplemented and amended and the Environmental Report as supplemented and amended.
 - B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:
 - (1) PSEG Nuclear LLC, and Constellation Energy Generation, LLC to possess the facility at the designated location in Salem County, New Jersey, in accordance with the procedures and limitations set forth in this renewed license;
 - (2) PSEG Nuclear LLC, pursuant to Section 104b of the Act and 10 CFR Part 50, "licensing of Production and Utilization Facilities," to possess, use and operate the facility;

- (3) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;
 - (4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
 - (5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;
 - (6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Mechanical disassembly of the Westinghouse Cobalt Burnable Absorber (COBA) Assemblies containing Cobalt-60 is not considered separation; and
 - (7) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, and transfer Cobalt-60.
- C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:
- (1) Maximum Power Level
PSEG Nuclear LLC is authorized to operate the facility at a steady state reactor core power level not in excess of 3459 megawatts (one hundred percent of rated core power).
 - (2) Technical Specifications and Environmental Protection Plan
The Technical Specifications contained in Appendix A, as revised through Amendment No. 357, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications, and the Environmental Protection Plan.
 - (3) Deleted Per Amendment 22, 11-20-79

(4) Less than Four Loop Operation

PSEG Nuclear LLC shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of Appendix A to this renewed license) with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensees and approval for less than four loop operation at power levels above P-7 has been granted by the Commission by Amendment of this renewed license.

- (5) PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, subject to the following provision:

PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

- (6) The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;
2. Identification of the procedures used to measure the values of the critical parameters;
3. Identification of process sampling points;
4. Procedure for recording and management of data;
5. Procedures defining corrective actions for off control point chemistry conditions; and
6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

(7) Systems Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic inspection requirements, and
2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC. Changes to the withdrawal schedule or storage requirements shall be submitted to the NRC as a report in accordance with 10 CFR 50.4.

- (21) PSEG Nuclear LLC shall take one core sample in the Unit 1 spent fuel pool west wall, by the end of 2013, and one core sample in the east wall where there have been indications of borated water ingress through the concrete, by the end of 2015. The core samples (east and west walls) will expose the rebar, which will be examined for signs of corrosion. Any sample showing signs of concrete degradation and/or rebar corrosion will be entered into the licensee's corrective action program for further evaluation. PSEG Nuclear LLC shall submit a report in accordance with 10 CFR 50.4 no later than three months after each sample is taken on the results, recommendations, and any additional planned actions.
- (22) Concurrent with the first use of the chilled water cross-tie as allowed by Technical Specification 3.7.10c, PSEG shall confirm the required performance of the chilled water system cross-tie.
- (23) PSEG is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 SSCs using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in the PSEG submittal letter dated January 31, 2025, and all its subsequent associated supplements as specified in License Amendment No. 353 dated February 20, 2026.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic PRA approach).

- (24) Prior to off-load of fuel assemblies containing irradiated COBA fuel inserts, verify that the incident energy flux on the Spent Fuel Pool walls is less than 10^{10} MeV/cm²-sec per NUREG/CR-6927 with the associated fuel assemblies placed at the targeted SFP rack locations.

D. Paragraph 2.D. has been combined with paragraph 2.E. per Amendment No. 86, June 27, 1988.

DESIGN FEATURES

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DESIGN PRESSURE AND TEMPERATURE

- 5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 47 psig. Containment air temperatures up to 351.3°F are acceptable providing the containment pressure is in accordance with that described in the UFSAR.

5.3 REACTOR CORE

FUEL ASSEMBLIES

- 5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO® or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

- 5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

COBALT BURNABLE ABSORBER (COBA) ASSEMBLIES

- 5.3.3 Salem is authorized to place a maximum of 288 cobalt burnable absorber rodlets in an operating cycle.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

- 5.4.1 The reactor coolant system is designed and shall be maintained:



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

PSEG NUCLEAR LLC

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 339
Renewed License No. DPR-75

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Constellation Energy Generation, LLC (the licensees) dated September 24, 2025, as supplemented by letters dated June 24, 2026, and August 31, 2026, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to license conditions and technical specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 339, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Wrona, Chief
Operating Reactor Licensing and Data Branch 1
Division of Licensing Projects I
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed
Facility Operating License
and Technical Specifications

Date of Issuance: September 29, 2026

ATTACHMENT TO LICENSE AMENDMENT NO. 339
SALEM NUCLEAR GENERATING STATION, UNIT NO. 2
RENEWED FACILITY OPERATING LICENSE NO. DPR-75
DOCKET NO. 50-311

Replace the following pages of Renewed Facility Operating License No. DPR-75 with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

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Replace the following page of the Appendix A, Technical Specifications, with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

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- G. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;
 - H. The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public;
 - I. After weighing the environmental, economic, technical and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Renewed Facility Operating License No. DPR-75 subject to the conditions for protection of the environment set forth herein is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied;
 - J. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70;
 - K. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21 (c), such that there is reasonable assurance that the activities authorized by this renewed operating license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations; and
 - L. The production, possession, and transfer of Cobalt-60 as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30. This type of radioactive material is considered to be bulk material (ML25100A186).
2. Pursuant to approval by the Nuclear Regulatory Commission at meetings on January 14, 1981, April 28, 1981, and May 19, 1981, the License for Fuel-Loading and Low-Power Testing issued on April 18, 1980 is superseded by Renewed Facility Operating License No. DPR-75 hereby issued to PSEG Nuclear LLC, and Constellation Energy Generation, LLC (the licensees), to read as follows:
- A. This renewed license applies to the Salem Nuclear Generating Station, Unit No. 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located on the southern end of Artificial Island on the east bank of the Delaware River in Lower Alloways Creek Township in Salem County, New Jersey and is described in the Final Safety Analysis Report as supplemented and amended and the Environmental Report as supplemented and amended.
 - B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:
 - (1) PSEG Nuclear LLC, and Constellation Energy Generation, LLC to possess the facility at the designated location in Salem County, New Jersey, in accordance with the procedures and limitations set forth in the renewed license;

- (2) PSEG Nuclear LLC, pursuant to Section 104b of the Act and 10 CFR part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use and operate the facility at the designated location in Salem County, New Jersey, in accordance with the limitations set forth in this renewed license;
 - (3) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;
 - (4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration and as fission detectors in amounts as required;
 - (5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;
 - (6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Mechanical disassembly of the Westinghouse Cobalt Burnable Absorber (COBA) Assemblies containing Cobalt-60 is not considered separation; and
 - (7) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, and transfer Cobalt-60.
- C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:
- (1) Maximum Power Level
PSEG Nuclear LLC is authorized to operate the facility at steady state reactor core power levels not in excess of 3459 megawatts (thermal).
 - (2) Technical Specifications and Environmental Protection Plan
The Technical Specifications contained in Appendix A, as revised through Amendment No. 339, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in the PSEG submittal letter dated January 31, 2025, and all its subsequent associated supplements as specified in License Amendment No. 335 dated February 20, 2026.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic PRA approach).

- (39) Prior to off-load of fuel assemblies containing irradiated COBA fuel inserts, verify that the incident energy flux on the Spent Fuel Pool walls is less than 10^{10} MeV/cm²-sec per NUREG/CR-6927 with the associated fuel assemblies placed at the targeted SFP rack locations.

D. An exemption from certain requirements of Appendix J to 10 CFR Part 50 is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 4. This exemption was authorized by law and will not endanger life of property or the common defense and security and is otherwise in the public interest. The exemption, therefore, remains in effect. The granting of the exemption was authorized with the issuance of the License for Fuel-Loading and Low-Power Testing, dated April 18, 1980. The facility will operate, to the extent authorized herein, in conformity with the application as amended, the provisions of the Act, and the regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, submitted by letter dated May 19, 2006, are entitled: "Salem-Hope Creek Nuclear Generating Station Security Plan," "Salem-Hope Creek Nuclear Generating Station Security Training and Qualification Plan," and "Salem-Hope Creek Nuclear Generating Station Security Contingency Plan." The plans Contain Safeguards Information protected under 10 CFR 73.21.

PSEG Nuclear LLC shall fully implement and maintain in effect all provisions of the Commission-approved Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Salem-Hope Creek CSP was approved by License Amendment No. 283 as supplemented by changes approved by License Amendment Nos. 285, 288 and 299.

F. A temporary exemption from General Design Criterion 57 found in Appendix A to 10 CFR Part 50 is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 5, Section 6.2.3.1. This Exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. The exemption, therefore, is hereby granted and shall remain in effect through the first refueling outage as discussed in Section 6.2.3.1 of Supplement 5 to the Safety Evaluation Report. The granting of the exemption is authorized with the issuance of the Facility Operating License, dated May 20, 1981. The facility will operate, to the extent authorized herein, in conformity with the application as amended, the provisions of the Act, and the regulations of the Commission.

DESIGN FEATURES

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DESIGN PRESSURE AND TEMPERATURE

- 5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 47 psig. Containment air temperatures up to 351.3°F are acceptable providing the containment pressure is in accordance with that described in the UFSAR.

5.3 REACTOR CORE

FUEL ASSEMBLIES

- 5.3.1 The reactor core shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO® or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

- 5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

COBALT BURNABLE ABSORBER (COBA) ASSEMBLIES

- 5.3.3 Salem is authorized to place a maximum of 288 cobalt burnable absorber rodlets in an operating cycle.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

- 5.4.1 The reactor coolant system is designed and shall be maintained:
- In accordance with the code requirement specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,
 - For a pressure of 2485 psig, and
 - For a temperature of 650°F, except for the pressurizer which is 680°F.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR
AMENDMENT NO. 357 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-70
AMENDMENT NO. 339 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-75
PSEG NUCLEAR LLC
CONSTELLATION ENERGY GENERATION, LLC
SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

1.1 Application and Noticing Information

By letter LR-N25-0074 dated September 24, 2025 (Reference 1), as supplemented by letters LR-N26-0045 dated June 24, 2026 (Reference 2), and LR-N26-0052 dated August 31, 2026 (Reference 3), PSEG Nuclear LLC (the licensee) submitted license amendment request (LAR) S25-01 to request changes to the renewed facility operating licenses and technical specifications (TS) for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2.

A notice of consideration of proposed issuance of amendments with a proposed no significant hazards consideration (NSHC) determination was published in the *Federal Register* on February 24, 2026 (Reference 4), and there has been no public comment on the proposed NSHC determination. The supplements dated June 24, and August 31, 2026, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's proposed NSHC determination.

1.2 Proposed Changes

The licensee proposed revising the renewed facility operating licenses, including the appendices containing the TS, to allow irradiation of cobalt-59 (Co-59) and the production and transfer of cobalt-60 (Co-60). In its supplement dated August 31, 2026, the licensee stated that its analyses were performed at an increased power level in anticipation of an expected future power uprate; however, the licensee would operate Salem, Units 1 and 2 at the current licensed thermal power level of 3,459 megawatts thermal (MWt) with cobalt burnable absorber (COBA)

rodlets loaded in the core unless a future licensing action approves Co-60 generation at a higher power level.

To allow for the production and transfer of Co-60 in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," the licensee proposed new license conditions 1.K for Salem, Unit No. 1 and 1.L for Salem, Unit No. 2, which would state, "The production, possession, and transfer of Cobalt-60 as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30. This type of radioactive material is considered to be bulk material (ML25100A186)."

To clarify the term "separation" relative to the removal from the COBAs of enclosed capsules containing Co-60 after discharge from a Salem reactor core, the licensee proposed adding the following sentence to license condition 2.B.(6) for both units: "Mechanical disassembly of the Westinghouse Cobalt Burnable Absorber (COBA) Assemblies containing Cobalt-60 is not considered separation."

To allow intentional production of Co-60 during operation of the Salem units, the licensee proposed new license condition 2.B.(7) for both units, which would state, "PSEG Nuclear LLC, pursuant to the Act [Atomic Energy Act of 1954, as amended] and 10 CFR Part 30, to intentionally produce, possess, and transfer Cobalt-60."

To address concerns with the potential effects on spent fuel pool (SFP) wall integrity from the irradiated COBAs' gamma heating effects, the licensee proposed new license conditions 2.C.(24) for Salem, Unit No. 1 and 2.C.(39) for Salem, Unit No. 2, which would state:

Prior to off-load of fuel assemblies containing irradiated COBA fuel inserts, verify that the incident energy flux on the SFP walls is less than 10^{10} MeV/cm²-sec [megaelectron volts per square centimeter-seconds] per NUREG/CR-6927 with the associated fuel assemblies placed at the targeted SFP rack locations.

The licensee also proposed new TS 5.3.3, "Cobalt Burnable Absorber (COBA) Assemblies," which would state, "Salem is authorized to place a maximum of 288 cobalt burnable absorber rodlets in an operating cycle."

1.3 Background

In its LAR, the licensee stated that it is collaborating with Westinghouse Electric Company LLC (Westinghouse) and Nordion (Canada) Inc. (Nordion) to develop and implement a pilot program for producing Co-60 in the Salem reactors during power operation. The Co-60 would be intended for industrial and medical purposes.

In section 3.1, "Background and General Approach," of the LAR enclosure, "Evaluation of the Proposed Changes," the licensee stated that Co-60 would be produced by subjecting Co-59 material to neutron irradiation in the Salem units' reactor cores. Nordion would provide Westinghouse the Co-59 material, which would then be placed in metal capsules and loaded into fuel component assemblies, called COBA assemblies, at a Westinghouse facility. The COBAs would then be inserted into Westinghouse fuel assemblies, shipped to Salem, and loaded into the reactor cores as part of the fuel assembly where they would be irradiated for

multiple fuel cycles. When subjected to the neutron flux in the core, the Co-59 would be converted to Co-60.

In section 3.1 of the LAR enclosure, the licensee stated that after removal from the core, the activated capsules would be harvested (i.e., removed from the COBA rodlets), loaded into a Nordion-supplied, Canadian Nuclear Safety Commission (CNSC)-certified, and U.S. Department of Transportation-approved transport package, and transferred to Nordion's Canada facility. At Nordion, the Co-60 slugs would be removed from the Westinghouse irradiated capsules, and Nordion would create finished sources to meet various end-use specifications.

In section 3.2, "Plant Implementation," of the LAR enclosure, the licensee stated that it plans to initially load eight COBAs in host fuel assemblies during the Salem, Unit No. 1 refueling outage occurring in fall 2026. Each COBA would be inserted into a fresh fuel assembly and loaded into the reactor core with its host fuel assembly. The licensee stated that the fuel assembly itself is not modified. This process would continue at each subsequent refueling outage with irradiated COBAs shuffled into fresh fuel assemblies and additional unirradiated COBAs loaded into fresh fuel assemblies. After the requisite number of activation cycles in the core, the COBAs would be moved to the cask pit for harvesting of the Co-60-containing capsules. During harvesting, each rodlet would be removed from the COBA baseplate, and each capsule would have its activity measured. These activity levels would be documented to assure that the total activity level of the byproduct material in each Nordion transport cask is maintained within the licensed activity limit of the cask certification.

2.0 REGULATORY EVALUATION

2.1 Requirements

Under 10 CFR 50.92(a), in determining whether an amendment to a license will be issued, the NRC staff is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. The common standards for licenses in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be "reasonable assurance" that the activities at issue will not endanger the health and safety of the public, and that the applicant will comply with the Commission's regulations.

The regulations in 10 CFR Part 20, "Standards for Protection Against Radiation," establish standards for protection against ionizing radiation resulting from activities conducted under licenses issued by the NRC.

Activities requiring a byproduct material license are covered under the requirements specified in 10 CFR 30.3, "Activities requiring license." The regulation in 10 CFR 30.3(a) states, except as noted in paragraph (a), "... no person shall manufacture, produce, transfer, receive, acquire, own, possess, or use byproduct material except as authorized in a specific or general license issued in accordance with the regulations in this chapter." Furthermore, 10 CFR Part 31, "General Domestic Licenses for Byproduct Material," section 31.1, "Purpose and scope," states, "This part establishes general licenses for the possession and use of byproduct material and a general license for ownership of byproduct material." Because either a specific or a general license is required and the provisions of the general license do not address production or transfer of byproduct material, a specific byproduct license is required. Therefore, the

requirement for a Part 30 byproduct material license is addressed by amendment to the existing 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," facility operating license.

The regulation in 10 CFR 50.36, "Technical specifications," requires that TSs include limiting conditions for operation (LCOs) and design features. The regulation in 10 CFR 50.36(c)(2) states, in part, that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility and requires that when a LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. The regulation in 10 CFR 50.36(c)(4) requires that the design features to be included in the TS are those features of the facility, such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in the safety limits, limiting safety systems settings, limiting control settings LCOs, and surveillance requirements sections of the TS.

The regulation in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires, in part, nuclear power reactors' emergency core cooling systems be designed so that their calculated cooling performance conforms to either the criteria in paragraph (b) of the regulation or to the features of 10 CFR, Part 50, Appendix K, "ECCS Evaluation Models." Chapter 15, "Accident Analysis," of the Salem UFSAR describes analyses to support compliance with 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

The regulation in 10 CFR 50.67, "Accident source term," sets limits for the radiological consequences of a postulated design basis accident (DBA) in terms of total effective dose equivalent (TEDE) using an alternative source term (AST). The NRC approved the full scope implementation of alternative source term methodology for Salem in 2006 (Reference 5).

2.2 Mandated Licensing Basis Document Information

Salem, Unit Nos. 1 and 2 were originally designed to comply with the Atomic Energy Commission (AEC) "Proposed General Design Criteria [GDC] for Nuclear Power Plant Construction Permits" published in July 1967 (AEC proposed GDC). Section 3.1.2, "Conformance with AEC Proposed General Design Criteria (July 1967)," of the Salem Updated Final Safety Analysis Report (UFSAR) (Reference 6) documents Salem's conformance with the AEC proposed GDC. The NRC staff reviewed the LAR against the following AEC proposed GDC, as described in the UFSAR:

- GDC 2, "Performance Standards," which states, in part, that those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects.
- GDC 6, "Reactor Core Design," which states that the reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected

conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power. The ability of the core to function throughout its lifetime without exceeding acceptable fuel damage limits is discussed in section 4 of the UFSAR. Detailed information on core design and performance is also included in section 4 of the UFSAR. Decay heat removal systems are discussed in sections 5, 6, 9, and 15 of the UFSAR. Section 15 of the UFSAR demonstrates that adequate fuel integrity is maintained under postulated abnormal situations.

- GDC 10, "Containment," which states that reactor containment shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain functional capability for as long as the situation requires.
- GDC 19, "Protection Systems Reliability," which states that protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.
- GDC 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention," which states, in part, that the reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures.
- GDC 35, "Reactor Coolant Pressure Boundary Brittle Fracture Prevention," which states that under conditions where reactor coolant pressure boundary systems' components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120 degrees Fahrenheit (°F) above the nil ductility transition temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation, or 60 °F above the nil ductility transition temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.
- GDC 41, "Engineered Safety Features Performance Capability," which states that engineered safety features such as containment heat removal system provides sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.
- GDC 42, "Engineered Safety Features Components Capability," which states that the capability of each component and system to perform its required function is not impaired by the effects of a LOCA.
- GDC 49, "Containment Design Basis," which states that the containment and its associated heat removal systems shall be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA, including considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

Chapter 15, "Accident Analysis," of the UFSAR describes safety margins for anticipated operational occurrences and DBAs, and the acceptance criteria and input assumptions used in transient and accident evaluations, including loss-of-coolant accident (LOCA) and non-LOCA events.

2.3 Precedents or Prior NRC Approvals

- Amendment Nos. 271 and 252 for Salem, Unit Nos. 1 and 2, regarding full scope implementation of alternative source term methodology (Reference 5)
- Amendment Nos. 354 and 336 for Salem, Unit Nos. 1 and 2, regarding the revised criticality safety analysis for the SFP and new fuel storage vault (NFSV) (Reference 7)
- Amendment No. 8 for Watts Bar Nuclear Plant, Unit 1, regarding lead test assemblies containing tritium-producing burnable absorber rods (Reference 8)
- Amendment Nos. 278 and 269 for Sequoyah Nuclear Plant, Units 1 and 2, regarding irradiation of tritium producing burnable absorber rods (Reference 9)

2.4 NRC-approved White Papers and Topical Reports

- Westinghouse Electric Company LLC, letter to NRC, "Submittal of the Westinghouse Cobalt-60 Digital Serialization Approach White Paper" (Reference 10), as supplemented (Reference 11). The NRC staff's review of this white paper is documented in its letter to Westinghouse dated July 23, 2025 (Reference 12).
- WCAP-9272(P)/9273(NP), "Westinghouse Reload Safety Evaluation Methodology" (Reference 13), which includes a copy of the NRC staff's approval
- NRC acceptance of WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version" (References 14 and 15)
- NRC acceptance of WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," (Reference 16)
- NRC acceptance of WCAP-15025-P-A, Revision 0, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17×17 Rod Bundles with Modified LPD Mixing Vane Grids" (Reference 17)
- NRC acceptance of WCAP-18240-P-A, "Westinghouse Thermal Design Procedure (WTDP)" (Reference 18)

2.5 NRC or NRC-endorsed Guidance

- Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors" (Reference 19), provides guidance on the potential susceptibility of recirculation sump screens to debris blockage during DBAs.

- NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants,” Chapter 4, “Reactor,” Section 4.2, Revision 3, “Fuel System Design” (Reference 20), provides guidance for the review of fuel system materials and designs.
- NUREG-1431, Revision 5, “Standard Technical Specifications, Westinghouse Plants,” Volume 1, “Specifications,” and Volume 2, “Bases” (Reference 21) provides guidance on the content of the standard technical specifications.
- NUREG/CR-6927, “Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures – A Review of Pertinent Factors” (Reference 22), provides guidance for reviewing the effects of irradiation on concrete structures.
- Regulatory Guide (RG) 1.28, Revision 6, “Quality Assurance Program Criteria (Design and Construction)” (Reference 23), provides guidance on quality assurance criteria for fuel assembly components and inserts
- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors” (Reference 24), provides guidance on acceptable applications of alternative source terms.

2.6 Codes and Standards

- American Concrete Institute (ACI) 349-01, “Code Requirements for Nuclear Safety Related Concrete Structures,” Appendix A, “Thermal Considerations,” 2001 (Reference 25)
- American National Standards Institute/American Nuclear Society (ANSI/ANS)-5.1-1979, “Decay Heat in Light Water Reactors,” August 1979 (Reference 26)
- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III, “Rules for Construction of Nuclear Facility Components,” Division 1, Subsection NG, “Core Support Structures,” Article NG-3000, 2023 Edition (Reference 27). Article NG-3000 addresses the structural qualification of core support structures, including rodlets, capsules, and mechanical connections subjected to vibration, seismic, and operational loads.
- International Organization for Standardization (ISO) 2919, “Radiological Protection — Sealed Radioactive Sources — General Requirements and Classification,” 2012 (Reference 28)

3.0 TECHNICAL EVALUATION

As discussed in section 2.1, under 10 CFR 50.92(a), in determining whether an amendment to a license will be issued, the NRC staff is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. The staff evaluated the request to determine whether the proposed changes are consistent with the regulations, licensing basis, and guidance discussed in section 2 of this safety evaluation. The staff reviewed the proposed changes to determine whether they meet applicable requirements and provide reasonable

assurance that operation with the proposed changes will not endanger the health and safety of the public in the technical areas of the mechanical design, thermal hydraulic design, materials, nuclear design and safety analyses, harvesting, measurement, marking and tracking, and transportation.

In attachment 2 (proprietary) to its LAR, the licensee provided the Westinghouse technical document, "Cobalt-60 Supplemental Information to Support the Use of Cobalt Burnable Absorber (COBA) Assemblies in Pressurized Water Reactors." Attachment 5 to the LAR is a nonproprietary version of attachment 2. In attachment 3 (proprietary), "Site Specific Information in Support of Cobalt Burnable Absorber (COBA) Assemblies in Salem Generating Station Units 1 and 2," to its LAR, the licensee provided site-specific material to complement the information provided in attachments 2 and 5 for certain topics. Attachment 6 to the LAR is a nonproprietary version of attachment 3. The NRC staff evaluated the information in attachments 2 and 5, as complemented by the information in attachments 3 and 6 and as supplemented by letters dated June 24, and August 31, 2026, only for applicability at Salem, Unit Nos. 1 and 2. This safety evaluation does not constitute NRC approval of the information in LAR, as supplemented, for generic applicability.

As discussed in its summary report (Reference 29), the NRC staff conducted a regulatory audit to support its review of the LAR.

3.1 Evaluation of Cobalt Burnable Absorber (COBA) Assembly Design

Section 2, "Description of the Cobalt-60 Production Process," and section 3, "Cobalt Burnable Absorber (COBA) Assembly Design," of attachment 2 to the LAR provide a high-level description of the COBA assembly design and fabrication. Section 2 of attachment 2 to the LAR provides overview of the COBA assembly design, and section 3 of attachment 2 to the LAR provides an overview of the assembly design considerations and fabrication process.

The NRC staff evaluated the COBA assembly design considerations and fabrication process described in sections 2 and 3 of attachment 2 to the LAR against AEC proposed GDC 6, as described in the UFSAR. GDC 6 states, in part, that the reactor core shall be designed to function without exceeding acceptable fuel damage limits and that the core design, with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated.

Section 2 of attachment 2 to the LAR states that the existing wet annular burnable absorber (WABA) platform is used as a design basis for the COBA assembly design. The WABA platform is a type of fuel assembly insert used to control reactivity within nuclear reactors. Section 4.2.3.1.3 of the UFSAR describes the use of the WABA platform at Salem. The NRC has approved other applications for assemblies that use the WABA platform for tritium-producing burnable absorber rod lead test assemblies (References 8 and 9). These precedents are relevant to the proposed COBA assembly design because both the tritium and Co-60 producing processes use similar fuel assembly inserts installed in the reactor core in the same manner as discrete burnable absorbers or other core components. Therefore, the NRC staff has determined that these precedents provide an acceptable framework for integrating the Co-60 producing technology into Salem's fuel assembly structure. The NRC staff has determined that

using the WABA platform as a design basis for the COBA assembly design conforms to AEC proposed GDC 6, as described in the UFSAR, and, therefore, is acceptable.

By selecting robust materials (i.e., ~~[[~~ ~~]]~~ implementing ~~[[~~ ~~]]~~ for the COBA assembly, the NRC staff determined that the COBA assembly design considerations and fabrication conform to AEC proposed GDC 6, as described in the UFSAR, and, therefore, are acceptable.

3.2 Evaluation of the COBA Mechanical Design

Section 4, "Mechanical Design Evaluation," of attachment 2 to the LAR describes the COBA design criteria; the cobalt slug and capsule configuration, mechanical integrity, creep analysis, production leak test, and wear evaluation; and COBA baseplate connections. Section 4 of attachment 2 to the LAR also provides an assessment of COBA insertion on operation risk and capsule breach on reactor coolant chemistry. Section 3.3 "COBA Fabrication Process," of attachment 2 to the LAR describes the COBA fabrication process.

3.2.1 Evaluation of COBA Design Criteria

Section 4.1, "Design Criteria," of attachment 2 to the LAR lists the design criteria for the COBA design. The NRC staff evaluated these criteria against AEC proposed GDC 6, as described in the UFSAR, and against the ASME BPV Code, Section III, Division 1, Article NG-3000, which addresses qualification of core support structures for mechanical integrity. The NRC staff reviewed the list of proposed design criteria in LAR and determined they are consistent with GDC 6 and Article NG-3000 of the ASME BPV Code and, therefore, are acceptable.

3.2.2 Evaluation of Cobalt Slug/Capsule Configuration

Section 4.2, "Cobalt Slug/Capsule Configuration," of attachment 2 to the LAR describes the dimensions and configuration of the cobalt slugs and capsules. Figure 2-1, "COBA Assembly," in attachment 2 to the LAR illustrates the configuration of the cobalt slugs and capsules. The NRC staff reviewed the cobalt slug and capsule configuration and confirmed that the design provides for maintaining proper clearances for the components and precluding improper mechanical interference during the COBA's service life. Therefore, the NRC staff determined that the proposed design configuration meets AEC proposed GDC 6, as described in the UFSAR, and, therefore, is acceptable.

3.2.3 Evaluation of Capsule/COBA Mechanical Integrity

Section 4.3, "Capsule/COBA Mechanical Integrity," of attachment 2 to the LAR describes design considerations and assumptions for maintaining the mechanical integrity of the COBAs and capsules under various loading conditions, including during operating conditions, shipping, handling, and seismic events. The NRC staff reviewed the licensee's assessment of COBA and capsule mechanical integrity under different loading conditions against Article NG-3000 of the ASME BPV Code, Section III, Division 1, and determined that the design considerations and assumptions for maintaining the mechanical integrity of the COBAs are consistent with this standard. The NRC staff determined that it has reasonable assurance that the proposed core components, assemblies, and connections will be qualified to withstand operational, seismic, and shipping/handling loads and, therefore, will maintain their mechanical integrity throughout

their service life. Therefore, the staff finds the design considerations and assumptions used for maintaining the mechanical integrity of the COBAs and capsules under various loading conditions are acceptable.

3.2.4 Evaluation of the Creep Analysis of the COBA Capsule

Section 4.4, "Creep Analysis of the COBA Capsule," of attachment 2 to the LAR, as supplemented by letter dated June 24, 2026, describes the creep analysis (i.e., the evaluation of a material's deformation under long-term conditions) of the COBA capsules for determining sufficient dimensional clearance for removal of the irradiated slugs from the capsules under expected operating conditions. The LAR, as supplemented, describes the capsules' materials and construction, conditions to which the capsules will be subjected, and the creep analysis criteria, assumptions, and inputs. As discussed in its audit summary report, the NRC staff audited the Westinghouse calculations supporting the creep analysis to confirm the inputs and results discussed in the LAR. The NRC staff reviewed the information in section 4.4 of attachment 2 to the LAR, as supplemented, against AEC proposed GDC 6, as described in the UFSAR. The NRC staff determined that the licensee's analysis is acceptable because [[

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Therefore, the NRC staff determined that the creep analysis described in section 4.4 of attachment to the LAR, as supplemented, conforms to AEC proposed GDC 6, as described in the UFSAR, and, therefore, is acceptable.

3.2.5 Evaluation of the Capsule Production Leak Test

Section 4.5, "Capsule Production Leak Test," of attachment 2 to the LAR describes the proposed leakage test to be performed on the capsules to ensure that the capsules are leak-tight and will not allow any coolant to enter the capsules. The licensee stated in the LAR that production capsules are required to pass a leak test using a [[
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The NRC staff evaluated the licensee's proposed approach for leak testing. Although no NRC guidance explicitly endorses this type of testing method, the method is widely used in NRC-approved fuel assembly fabrication processes, including fuel rod leak testing. The NRC staff determined that the [[
]] has high sensitivity and reliability in detecting leaks and provides reasonable assurance that capsules are leak-tight and coolant ingress will be prevented. The NRC staff's engineering judgment is based on the testing method's ability to detect extremely small leakage pathways, the method being widely used in nuclear industry quality assurance programs, and the COBA capsules' similarity to other components subject to this testing method.

Based on these considerations and the NRC's staff's experience with fuel-fabrication quality assurance programs, the NRC staff determined that the proposed testing method would provide high sensitivity and reliability for detecting capsule leakage. Therefore, NRC staff finds that the proposed capsule production leak testing is acceptable.

3.2.6 Evaluation of Capsule Wear Analysis

Section 4.6, "Capsule Wear Evaluation," of attachment 2 to the LAR describes the licensee's analysis of capsule wear, including the results of a prototype wear test featuring a production COBA rodlet in a typical Westinghouse thimble tube, for determining the vibration-induced wear of the COBA capsules (and slugs within them) against surrounding components. The purpose of the capsule wear analysis was to confirm that the COBA capsules and internal slugs maintain structural integrity and do not experience unacceptable wear throughout their service life under reactor operating conditions. The licensee's testing concluded that [[

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The NRC staff assessed the wear analysis and results against section 4.2 of the SRP; the ASME BPV Code, Section III, Division 1, Article NG-3000; and RG 1.28 to determine if the licensee's wear analysis and results provide reasonable assurance that integrity and reliability of the reactor components will be maintained. The NRC also audited documentation associated with the wear evaluations to confirm the results stated in section 4.6 of attachment 2 to the LAR. The NRC staff confirmed that the wear experienced by COBA capsules and slugs during testing remained within the testing acceptance criteria. The NRC staff determined that the capsule wear evaluation provides reasonable assurance that the COBA capsules and internal slugs will not experience unacceptable wear throughout their service life and, therefore, will maintain structural integrity. Therefore, NRC staff finds that the wear analysis meets the regulatory criteria cited above, and, therefore, is acceptable.

3.2.7 Evaluation of Capsule Integrity for Shipment ISO 2919 Testing

Section 4.7, "ISO 2919 Testing," of attachment 2 to the LAR describes how COBA capsule structural integrity and radiological safety is tested prior to shipment and the results of prototype testing. The transport package, certified by the Canadian Nuclear Safety Commission (CNSC) and endorsed by the U.S. Department of Transportation (DOT), mandates compliance with ISO 2919.

The evaluation of capsule integrity during shipment is grounded in three primary standards and certifications:

- ISO 2919, which specifies general requirements and classification for sealed radioactive sources, including mandatory mechanical and environmental testing
- CNSC certification, which confirms the transport package meets Canadian regulatory expectations for safety and integrity
- DOT endorsement, which ensures compliance with U.S. transportation regulations for radioactive materials.

ISO 2919 specifies the following tests to validate capsule durability and safety:

- Bending, puncture, and impact tests. These tests are conducted on empty capsule tubing, representing a conservative scenario by excluding structural support from welded end caps and cobalt slugs.

- Vibration, pressure, and temperature tests. These tests are performed on completed capsules, as the slugs and end caps may influence the results. The licensee stated that the capsules are visually inspected after the tests are completed.

Section 4.7 of attachment 2 to the LAR stated that successful prototype testing has been performed on all thicknesses of tube wall, including two that were thinner than the final capsule design, and that no defects were observed. The LAR also states that testing of the production capsule design will also be performed. The NRC staff determined that the absence of observed defects and the successful passage of all ISO 2919 tests would confirm that the capsules possess the structural integrity necessary to withstand the mechanical and environmental stresses encountered during shipment.

3.2.8 Evaluation of the Baseplate Connections

Section 4.8, "Baseplate Connections," of attachment 2 to the LAR describes the purpose, function, and operation involving the baseplate, which forms the structural base for the COBA insert and maintains the lateral and vertical position of the rodlets. The LAR states that pull and bending tests were performed on the baseplate connection to confirm that the connection will withstand all expected handling loads. The NRC staff reviewed the LAR's description of the baseplate and determined that the baseplate purpose, function, and operation provide reasonable assurance of the structural integrity of the COBA assembly. Therefore, the NRC finds the baseplate design is acceptable.

3.2.9 Evaluation of COBA Insertion on Operation Risk

Section 4.9, "Assessment of COBA Insertion on Operation Risk," of attachment 2 to the LAR, as supplemented by letter dated June 24, 2026, describes the licensee's analyses, which includes a failure modes and effects analysis (FMEA), for ensuring the integrity of the capsules in preventing cobalt ingress into the reactor coolant. Table 4.9-1, "Summary Assessment of COBA Capsule Failure Modes Resulting in Capsule Breach," of attachment 2 to the LAR identifies potential mechanisms for capsule breach and how they were dispositioned. The LAR states that all failure mechanisms were judged to be of low likelihood and would not be expected to result in capsule failure given the capsule operating environment and residence time. The LAR provided the overall probability of capsule failure and likelihood of cobalt release from the addition of COBAs into the core.

The NRC staff audited the calculations and documentation associated with the FMEA, specifically regarding capsule breach caused by **[[** **]]** to confirm the conclusions provided in the LAR. The NRC staff determined that the assessment described in the LAR provides reasonable assurance of capsule integrity and low risk of radiological release. Therefore, the NRC staff finds the assessment of COBA insertion on operation risk acceptable.

3.2.10 Evaluation of Impact of COBA Capsule Breach on Reactor Coolant Chemistry

Section 4.10, "Impact of COBA Capsule Breach on Reactor Coolant Chemistry," in attachments 2 and 3 of the LAR describes the impact of COBA capsule breach on reactor coolant chemistry and include results of studies by Westinghouse to estimate the rate of cobalt release into the reactor coolant system (RCS) in the event of a capsule breach. The NRC staff's evaluation of this information is in section 3.6.9 of this safety evaluation.

3.3 Evaluation of Thermal and Hydraulic Assessment

Section 5, "Thermal and Hydraulic Evaluation," of attachment 2 to the LAR, as supplemented by letter dated August 31, 2026, describes the licensee's assessment of the thermal and hydraulic impacts of the proposed changes, including an assessment of departure from nucleate boiling (DNB), cooling and forces experienced by reactor core components, core bypass flow and control rod function, and the thermal-hydraulic analysis of the COBA rodlet design.

3.3.1 Evaluation of DNB and Core Components

In section 5.1, "DNB and Core Components," of attachment 2 to the LAR, the licensee assessed three potential impacts associated with the proposed insertion of COBAs: cooling of the COBAs and thimble tubes, impacts on DNB, and COBA capsule lift force.

Cooling of COBAs and Thimble Tubes

In section 5.1 of attachment 2 to the LAR, as supplemented by letter dated August 31, 2026, the licensee indicated that its assessment of the adequacy of the cooling of the COBAs and surrounding thimble tubes relies upon standard Westinghouse design methods typically used to evaluate cooling of fuel assembly components, such as control rods and WABAs, and augmented techniques that include computational fluid dynamics (CFD) analysis.

The licensee established an acceptance criterion that bulk boiling should be precluded [(] The licensee used CFD analysis to determine coolant temperatures within and around the COBAs to assess [(] The CFD analysis relied upon boundary conditions determined by standard Westinghouse nuclear and thermal-hydraulic design methods. The licensee further relied upon the coolant flows and temperatures calculated by CFD analysis to assure acceptable COBA component temperatures.

The NRC staff reviewed the licensee's approach for assessing the cooling of the COBAs and thimble tubes and determined that the licensee appropriately applied standard design methods to the extent applicable in support of the CFD analysis of the assessment of cooling to the COBAs and thimble tubes. The NRC staff's evaluation of the licensee's CFD analysis, [(] and associated considerations is provided in section 3.3.3 of this safety evaluation.

DNB Impacts

Regarding DNB, in section 5.1 of attachment 2 to the LAR the licensee indicated that impacts on a hot assembly containing COBA inserts would be similar to impacts associated with existing fuel assembly inserts (e.g., the use of WABAs as secondary burnable absorbers in addition to integral fuel burnable absorbers). The licensee further indicated that DNB impacts are expected to be minor and within the range of typical cycle-to-cycle reloads. The licensee further noted that the gamma energy released by the activated Co-60 would be negligible relative to the heat generated by fuel rods for the purpose of computing DNB.

The NRC staff finds the licensee's approach for assessing DNB described in section 5.1 of attachment 2 to the LAR to be reasonable. While the NRC staff expects the heat generated by the activated Co-60 in the COBAs to exceed those of typical inserts, such as WABAs, the

generated heat is expected to be small relative to the power of an idealized hot fuel rod assessed for DNB performance in the plant safety analysis. The NRC staff's evaluation of the impact of COBAs on the specific design-basis transients and accidents for which DNB may be among the acceptance criteria is provided in section 3.6.3 of this safety evaluation.

COBA Capsule Lift Force

In the "Audit Item 13" section of the supplement dated August 31, 2026, the licensee indicated that it relied upon inputs derived from a [[]]] and CFD analyses to calculate the force necessary to lift cobalt capsules within a COBA rodlet outer tube, considering [[]]] As discussed in section 5.1 of

attachment 2 to the LAR, [[]]]

]] the licensee determined that the COBA capsules will not experience liftoff during operation because the maximum calculated lift force is less than the weight of the capsules.

The NRC staff audited the licensee's supporting analyses during its review of the LAR. In the "Audit Item 13" section of its supplement dated August 31, 2026, the licensee provided its engineering rationale and an illustrative calculation to demonstrate the negligible impact of the flow paths omitted from its analyses. In its supplement, the licensee further quantified [[]]]

]] Based upon its review of the information in the licensee's application, as supplemented, the NRC staff determined that the licensee's assessment of COBA capsule lift force is acceptable.

3.3.2 Evaluation of Core Bypass Flow and Control Rod Function

In section 5.2, "Core Bypass Flow and Control Rod Function," of attachment 2 to the LAR, the licensee discussed its evaluation of the potential hydraulic impacts resulting from the insertion of COBAs on reactor vessel internal components, including its validation of the impact of the COBA inserts on the following assumptions in its non-LOCA analysis discussed in section 8.3, "Non-LOCA Transients," of attachment 2 to the LAR:

- no adverse impacts on the core bypass flow fraction and core pressure drop
- no adverse impacts on rod drop time
- no adverse impacts on the reactor trip rod position versus time curve

Regarding the first assumption, the licensee indicated in section 5.2 of attachment 2 to the LAR that it is relying upon the THRIVE computer code to determine various hydraulic flow parameters within the reactor pressure vessel, including pressure losses and core bypass flows. The licensee indicated that its THRIVE calculations determined that the presence of COBA inserts would not impact the total design core bypass flow. Regarding the latter two assumptions, the licensee stated that it is relying upon the DROP computer code to calculate the time required for control rods to drop into the core. The licensee indicated that because there are no changes to fuel assembly flow areas, thimble tube dimensions, nor core bypass flow, the rod-drop behavior determined by the DROP code would not be impacted.

The NRC staff finds the use of standard vendor design methods, including use of the DROP and THRIVE codes, acceptable for determining the bypass flow fraction and pressure drop because: (1) these methods have been demonstrated to be appropriate for calculating external flow around the COBAs based upon their history of application to similar fuel assembly inserts, and (2) the limited number of COBAs the licensee has proposed to insert would have limited influence on the overall bypass flow and pressure drop balance in the core. The NRC staff also finds the licensee's rationale that control rod drop times would not be affected by COBA insertion to be acceptable because of the COBA's lack of impact on the set of parameters governing control rod drop. Therefore, the NRC staff finds the licensee's assessment of COBA insertion on core bypass flow and control rod function to be acceptable.

3.3.3 Evaluation of COBA Rodlet Design Thermal-Hydraulic Analysis

3.3.3.1 Summary of Licensee's Assessment and Results

In section 5.3.1, "Design Criteria," of attachment 2 to the LAR, the licensee listed five thermal-hydraulic design criteria for the COBA rodlets. The licensee stated that it followed standard design methodologies for Westinghouse core components to demonstrate meeting the five thermal-hydraulic design criteria. A summary of the licensee's approach and the NRC staff's evaluation thereof is provided in sections 3.3.1 and 3.3.2 of this safety evaluation.

The licensee also established and evaluated a sixth thermal-hydraulic design criterion (i.e., that bulk boiling shall not occur [[

]]) This design criterion was treated separately from the other five criteria because the licensee relied upon CFD analysis rather than standard thermal-hydraulic design methodologies to perform its associated assessment. The licensee used CFD analysis to model the flow and heat transfer characteristics of [[

]]

In section 5.3.1 of attachment 2 to the LAR, the licensee indicated that the CFD model predicted no boiling in all flow paths within the COBA rodlets and thimble tubes with considerable margin. The licensee also indicated that the CFD analysis modeled the fluid flows and temperatures [[

]]

Figure 1 below further illustrates the physical geometry and key cooling flow paths. [[

]]

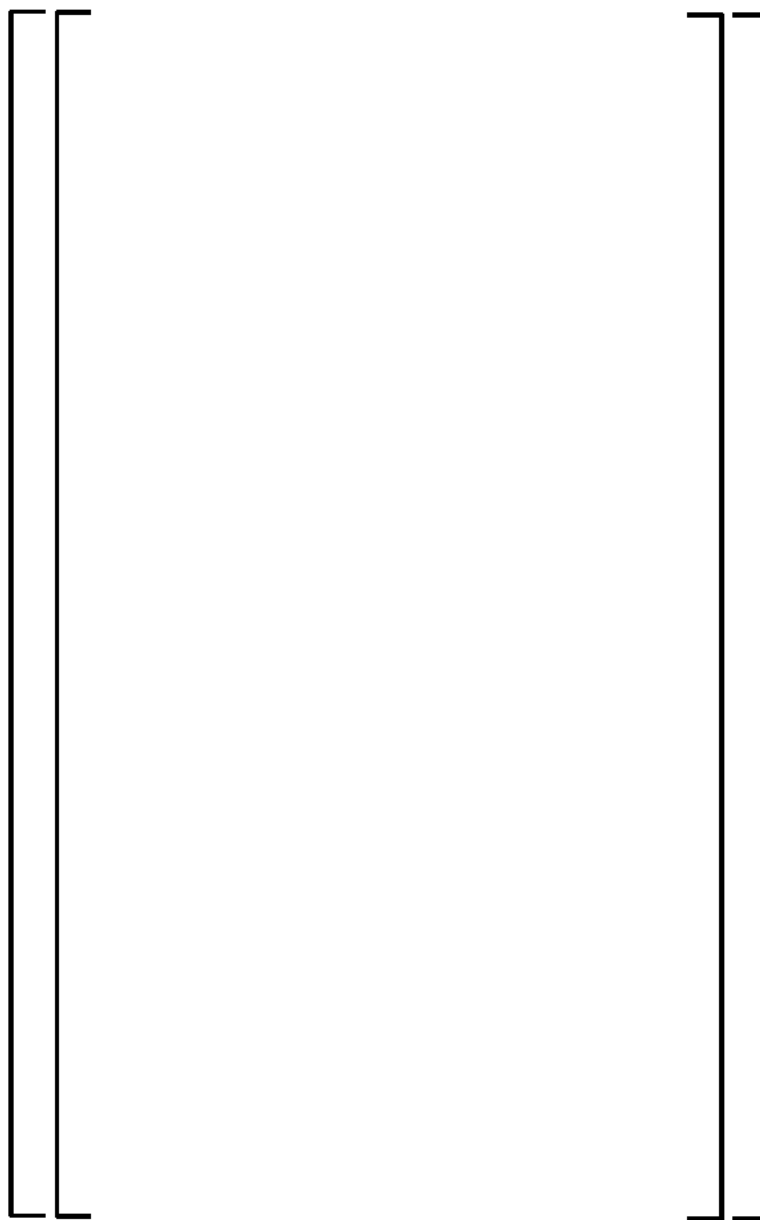


Figure 1: [[

]]

In the “Audit Item 2” section of its supplement dated August 31, 2026, the licensee identified that the location of the [[

]] A

qualitative schematic depicting this behavior is provided below in Figure 2, which zooms in on the upper portion of the diagram in Figure 1.



Figure 2: [[

]]

The licensee's analysis resulted in a [[
]] as described in the licensee's supplement dated August 31, 2026.

3.3.3.2 Evaluation of CFD Methodology

The NRC staff reviewed the LAR, as supplemented, and audited the licensee's CFD analysis and other supporting calculations to verify the licensee's conclusion in the LAR that there would be adequate [[
]]. Considering the [[
]] predicted by the licensee, the NRC staff focused its review on the licensee's CFD methodology, supporting validation for the licensee's CFD methodology, and the conservatism and uncertainties associated with the determination of the [[
]]

In section 5.3.1 of attachment 2 to the LAR, as supplemented, the licensee described the use of CFD analysis to demonstrate that no bulk boiling occurs in any flow path of the COBA rodlet and thimble tube configuration during normal operation, with margin to address limiting conditions expected during normal operation. The primary objective of the CFD analysis is to confirm that the [[

]] and that all acceptance criteria for thermal-hydraulic performance are satisfied. As stated in the "Audit Item 2" section of its supplement dated August 31, 2026, the licensee's CFD approach [[

]]

The licensee performed mesh sensitivity studies and turbulence model comparisons,

[[

]]

The validated CFD model, when applied with bounding plant conditions, predicts no bulk boiling in any flow path of the COBA rodlet/thimble tube configuration, [[

]] as described in the licensee's supplement dated August 31, 2026.

The NRC staff reviewed the licensee's approach to uncertainty quantification and conservatism,

[[

]]

These practices ensure that the CFD predictions are conservative with respect to both hydraulic resistance and thermal margin.

Based on its review, the NRC staff finds that the licensee's CFD methodology is consistent with industry best practices for complex internal flow and heat transfer problems, and that the modeling approach is appropriate for the evaluation of the COBA rodlet assembly under the

conditions analyzed. The staff notes that the licensee's use of conservative assumptions and sensitivity studies provides additional confidence in the robustness of the CFD predictions.

3.3.3.3 Evaluation of Thermal-Hydraulic Analysis Benchmarking

Section 5.3.2 of attachment 2 to the LAR, as supplemented by letter dated August 31, 2026, describes the licensee's benchmark testing performed to establish the credibility of the CFD methodology for the COBA rodlet design. The licensee's benchmarking and validation program incorporated a combination of [[
]]

[[
]]

The "Audit Item 2" section of the supplement dated August 31, 2026, describes the licensee's
[[

]]

[[
]]

The "Audit Item 2" section of the supplement dated August 31, 2026, describes the licensee's
[[

]]

The NRC staff reviewed the licensee's benchmarking program and determined that the [[
]] provides a robust and
representative basis for the application of the CFD methodology to the plant analysis. The NRC
staff determined that the use of [[
]] demonstrate
that the CFD model provides a conservative prediction of flow resistance in the COBA rodlet
assembly.

Plant CFD Model Application

The validated CFD methodology was applied to a detailed, plant-specific model of the COBA
rodlet assembly in the reactor core. The plant model was developed using ANSYS CFX
Version 24.2 and employed [[

]]

The NRC staff reviewed the licensee's plant-specific CFD analysis and determined that its
application to the COBA rodlet installation in the Salem cores is appropriate. The NRC staff
found that the use of [[

]] resulted in a technically sound and conservative representation of the limiting thermal-hydraulic behavior. The NRC staff further determined that output from the plant CFD model may be used as part of the licensee's demonstration of compliance with its proposed acceptance criteria for the COBA design, [[

]]

3.3.3.4 Assessment of [[]]

In light of the [[]]] calculated by the licensee, the NRC staff focused its review on relevant conservatisms and uncertainties associated with the [[]]] calculation. The NRC staff considered whether the additional refinements discussed in the licensee's supplement dated August 31, 2026, would provide reasonable assurance that the [[]]]

In its supplement dated August 31, 2026, the licensee provided additional information about the conservatisms and uncertainties associated with the determination of [[]]] that summarized relevant audit discussions and assessed key aspects of the calculation in four general areas: core power level, hydraulics, boundary conditions, and heat transfer. As described in its supplement dated August 31, 2026, the licensee's refined assessment included consideration of a number of individual factors under each general area and a quantification of the perceived conservatism or uncertainty associated with each factor. The licensee subsequently assessed the net effect of its assessment on the four general areas and concluded that [[]]] is assured.

The NRC staff performed a holistic review of the licensee's refined assessment provided in its supplement dated August 31, 2026. The NRC staff's review focused on whether the licensee's assessment provided reasonable assurance of [[]]] In its evaluation, the NRC staff relied on certain key aspects of the licensee's assessment but not others that were not necessary to confirm adequate [[]]] For example, some aspects from the licensee's assessment that the NRC staff did not rely upon in its determination include the following.

- The NRC staff did not rely upon the margin the licensee attributed to [[]]] One reason the NRC staff did not rely on flow margin is that the licensee had not assessed potential failure modes, time dynamics, and consequences associated with boiling in COBAs. In addition, [[]]] Therefore, the NRC staff determined that reliance upon the conservatively defined [[]]] provides an appropriate degree of protection to assure that boiling in COBAs is prevented. As such, the NRC staff did not rely upon [[]]] in the margin assessment.
- The NRC staff did not rely upon the margin the licensee [[]]] While the licensee provided plant-specific data spanning multiple operating cycles, section B.3.2.2 of volume 2 of NUREG-1431 (Reference 21) indicates that measurements of the nuclear enthalpy rise hot channel

factor are generally taken with the core at or near steady-state conditions. Because of the [[

]] the NRC staff's reasonable assurance finding did not rely upon the [[]]

- The licensee had not estimated the uncertainty associated with the [[

]]

Some of the key aspects of the licensee's assessment that the NRC staff determined to have supported a reasonable assurance finding of [[]] included the following.

- The licensee's CFD analysis was performed for an uprated power level of 3,708 MWt, which is approximately 7.2 percent greater than the current licensed thermal power of 3,459 MWt. The licensee reasonably estimated that analyzing to the uprated power level has a conservative impact on the calculated [[]]
- The licensee's CFD analysis made slightly conservative assumptions or demonstrated negligible modeling impacts associated with the [[

]]

- The licensee found that its CFD methodology [[

]]

Based on its review of this information, the NRC staff found that, overall, the licensee's assessment of the conservatisms and uncertainties associated with the [[]] provided reasonable assurance that boiling would not occur at the currently licensed thermal power. Although the NRC staff did not credit certain aspects of the licensee's margin assessment in its determination, the staff found the licensee's overall assessment to have (1) considered the majority of key uncertainties, (2) reasonably estimated impacts of conservatisms and uncertainties, (3) generally used conservative modeling practices, and (4) provided a credible estimate of the net impact of the conservatisms and uncertainties on the calculated margin-to-boiling. Therefore, the NRC staff found the licensee's conclusion that boiling would not occur in the COBAs to be acceptable for the currently licensed thermal power level and other currently applicable operating conditions.

3.4 Evaluation of Materials

Section 6, "Materials," of attachment 2 to the LAR describes the materials used in the fabrication of the COBA assemblies and provides the following design criteria that the licensee used to evaluate the materials:

- Keep cobalt out of the coolant and the coolant outside of the capsule.
- Maintain the integrity of the nickel coating.

Section 6 of attachment 2 to the LAR also describes how the COBA assembly materials meet the two design criteria above. The NRC staff determined that meeting these criteria would provide reasonable assurance that AEC proposed GDC 6, as described in the UFSAR, would be met. The NRC staff reviewed the licensee's analyses of these materials provided in sections 6.1 through 6.4 of attachment 2 to the LAR against the above criteria. The NRC staff determined that the materials specifications, properties, and applications for the capsules, COBA rodlet, baseplate connection, and associated parts, and the nickel coating and cobalt diffusion properties provide reasonable assurance that the two criteria above and AEC proposed GDC 6, as described in the UFSAR, are met and, therefore, are acceptable.

3.5 Evaluation of Nuclear Design

Section 7, "Nuclear Design," of attachment 2 to the LAR describes the licensee's analysis of how the COBA inserts function in the reactor core and interact with other core components. Section 7.1, "Methodology," discusses the methodology for modeling the COBA inserts. Section 7.2, "Benchmarking of Nuclear Materials," discusses the benchmarking of the nuclear methods used to model the COBA inserts. Section 7.3, "Reactor Core Design Approach," discusses the reactor core design approach for maximizing Co-60 production, and section 7.4, "Implications on Reload Safety Evaluation," discusses the implications of the COBA inserts on the reload safety analysis.

3.5.1 Evaluation of COBA Modeling Methodology

Section 7.1 of attachment 2 to the LAR discusses the design models COBA inserts using the PARAGON lattice code to generate homogenized group cross sections, which are then passed through the NEXUS nuclear data methodology into the three-dimensional nodal simulator code, Advanced Nodal Code (ANC). The LAR also states that the COBA inserts can also be modeled using the PARAGON2 lattice code for the purpose of generating homogenized group cross-section for in-core modeling. The LAR indicates that the modeling showed the cobalt activation cross section resulting in Co-60 is 0.136 percent less than the total neutron absorption cross section of Co-59. The modeling also assumes that all Co-59 neutron capture yields to the Co-60 ground state.

The NRC staff determined that explicit geometric and axial representation of the COBA inserts and coupling to ANC via qualified methodologies are appropriate for safety-significant core physics predictions (i.e., reactivity and flux and power distributions). The documented treatment of the metastable branch as negligible is reasonable because of the 0.136 percent delta value stated in the LAR. The NRC staff determined that PARAGON, PARAGON2, and ANC are qualified tools with long-standing use in reload safety analyses and avoid unvalidated modeling paths. The NRC staff also determined that the modeling methodologies for the COBA inserts

conform to AEC proposed GDC 6, as described in the UFSAR. Therefore, the NRC staff finds that the licensee's methodology for evaluating nuclear design impacts from the COBA inserts is acceptable.

3.5.2 Evaluation of the Benchmarking of Nuclear Methods

Section 7.2 of attachment 2 to the LAR describes the benchmarking methods and results of the PARAGON, PARAGON2, and SERPENT2 codes to demonstrate whether those codes can model COBA fuel assemblies in the reactor core with respect to power and flux distributions, COBA assembly reactivity, and cobalt cross sections and depletion chains.

The NRC staff reviewed the licensee's benchmarking methods and results and determined that they provide high confidence in the licensee's ability to model absorption and depletion associated with the COBA assemblies for determining reactivity and source term predictions. The NRC staff determined that verification against SERPENT2 across multiple loadings and fuel rod sizes provides reasonable assurance of low method bias and the ability of PARAGON and PARAGON2 to predict reactivity and power distributions from the presence of COBA assemblies in the reactor core. The NRC staff finds that benchmarking of PARAGON, PARAGON2, and SERPENT2 demonstrated the ability to accurately model COBA to enable conformance with AEC proposed GDC 6, as described in the UFSAR, and, therefore, is acceptable.

3.5.3 Evaluation of Reactor Core Design Approach

Section 7.3 of attachment 2 to the LAR describes the reactor core conditions and operations factoring into maximizing Co-60 production. Such conditions and operation include loading patterns within the core, cycle lengths, depletion strategies, core operating conditions, and fuel lattice design features. The LAR indicated that the specific reactor core conditions and operations will be determined on an ongoing basis considering the market requirements for Co-60 production.

The NRC staff determined that the proposed reactor core design approach maintains energy and reactivity trajectories within normal cycle-to-cycle variation by [[

]] The NRC staff finds that proposed reactor core design approach conforms to AEC proposed GDC 6, as described in the UFSAR, and, therefore, is acceptable.

3.5.4 Evaluation of Implications on Reload Safety Evaluation

Section 7.4 of attachment 2 the LAR summarizes the licensee's conclusions about the implementation of COBA or the cobalt depletion chain on reactor core methodology, analyses, and operations. The licensee concluded in the LAR that implementing COBA (including the cobalt depletion chain additions) does not alter the methodologies used for PARAGON, PARAGON2, or ANC, nor requires changes to the reload safety evaluation (RSE) methodology. The licensee also concluded in the LAR that standard calculational and measurement uncertainties will continue to be applied to core designs implementing COBA.

Because the RSE methodology and uncertainty treatment are unchanged and reload safety analysis checklist items (i.e., key safety parameters applied as part of the RSE) are not expected to be significantly impacted, the NRC staff determined that the inclusion of COBA conforms to AEC proposed GDC 6, as described in the UFSAR, and, therefore, is acceptable.

3.6 Evaluation of Safety Analyses

Section 8, "Safety Analyses," of attachments 2 and 3 to the LAR, as supplemented, provides the licensee's assessment of the impact of Co-60 production on plant safety analyses and radiological consequences.

3.6.1 Evaluation of Design Parameters and Best Estimate Flows

In section 8.1, "Design Parameters and Best Estimate Flows," of attachments 2 and 3 to the LAR, the licensee refers to the analysis in section 5.1 of attachment 2 to the LAR in which the licensee evaluated the impact of COBAs on the design and best estimate core bypass flow on DNB. To model the flow and heat transfer characteristics of the COBA rod and thimble configuration, the licensee used standard methods and CFD analysis techniques, as discussed in section 5.3 of attachment 2 to the LAR. From the results of the analysis, the licensee determined that although the best estimate flows increase slightly with the implementation of COBAs, the revised design and best estimate core bypass flows remain bounded by the previously reported core flow limits. Therefore, the licensee determined that the COBA assemblies would not impact the analysis of record (AOR) design parameters and best estimate flows. Based on the NRC staff's evaluation on the impact of COBAs on the design core flow and core bypass flow presented in section 3.3 of this safety evaluation, the NRC staff determined that the licensee's assessment provides reasonable assurance that the COBAs conform to AEC proposed GDC 6, as described in the UFSAR, and, therefore, is acceptable.

3.6.2 Evaluation of Design Transients and Control Systems

In section 8.2, "Design Transients and Controls Systems," of attachment 2 to the LAR, the licensee provided its evaluation of the impact of implementing COBAs on design transients; control systems operability, margin to trip, and component sizing; cold overpressure mitigation system and low temperature overpressure protection system (COMS/LTOPS); and decay heat.

3.6.2.1 Evaluation of Design Transients

Section 8.2 of attachment 2 to the LAR, as supplemented by letter dated August 31, 2026, discusses the impact of the implementation of Co-60 production on design, or fluid systems, transients. This section of the LAR refers to analyses discussed in sections 5.1 and 7.4 of attachment 2 to the LAR. The design parameter impacts from COBA are discussed in section 8.1 of attachments 2 and 3 to the LAR. The licensee concluded that no fuel kinematic items are expected to be significantly impacted by the implementation of COBA assemblies and that there are no significant fluence changes expected that will impact the pressure-temperature limits. As discussed in sections 3.3.1 and 3.5.4 of this safety evaluation, the NRC staff determined that the analyses in sections 5.1 and 7.4 of attachment 2 to the LAR, as supplemented, are acceptable; therefore, the NRC staff has reasonable assurance that the fuel kinematic items and pressure-temperature limits will not be significantly impacted. In sections 3.6.3 through 3.6.8 of this safety evaluation, the NRC staff further evaluated the

specific design transients discussed in sections 8.3 through 8.8 of attachments 2 and 3 of the LAR.

3.6.2.2 Evaluation of Control Systems Operability, Margin to Trip, and Component Sizing

Section 8.2 of attachment 2 to the LAR, and the section, "Audit Item 35," in the LAR supplement dated August 31, 2026, describe the licensee's analysis of the Co-60 production implementation impacts on the following nuclear steam supply system (NSSS) control systems:

- rod control system
- pressurizer pressure control system
- pressurizer level control system
- steam dump control system
- steam generator level control system

In its supplement dated August 31, 2026, the licensee stated that these control systems are not affected by the insertion of COBA inserts in fuel assembly thimbles and that none of the system setpoints or logic for steady state operation or for design basis operational transients needs to be revised. Based on this assessment, the licensee determined no significant nuclear fuel kinematic, fluence, best estimate flow or core delta-pressure (ΔP) changes would invalidate any of the determinations in the current AOR for NSSS design transients, plant operability and margin to trip, or COMS/LTOPS. The licensee concluded that the COBA inserts would not impact any NSSS design transient determination. The licensee also stated that COBA fuel inserts are burnable poisons similar to WABA fuel inserts but with different neutron absorbing material and a longer lifetime. The licensee stated that its no-impact assessment of the COBA fuel inserts on control systems is consistent with the no-impact determination on control systems of WABA inserts operating experience.

The NRC staff reviewed the licensee's assessment and determined that the COBA inserts would not affect the full power design conditions and would have negligible impact on the nuclear fuel kinetics parameters. The NRC staff confirmed that the control and protection system performance during steady state conditions, and during and following the design basis operational transients, and setpoints and time constants are not affected by the COBA inserts. Therefore, the NRC staff finds that the licensee's assessment provides reasonable assurance of conformance to AEC proposed GDC 19, as described in the UFSAR, and, therefore, is acceptable.

3.6.2.3 Evaluation of COMS/LTOPS

Section 8.2 of attachment 2 to the LAR describes why Co-60 implementation does not affect COMS/LTOPS. The licensee concluded that Co-60 implementation does not impact the design basis COMS/LTOPS transients or their mitigation strategies, nor the COMS/LTOPS setpoints. The NRC staff finds that because there are no significant changes to the best estimate RCS flows and core ΔP , the staff has reasonable assurance that the Co-60 implementation conforms to AEC proposed GDC 34 and 35 and, therefore, is acceptable.

3.6.2.4 Decay Heat from COBAs

Section 8.2 of attachment 2 to the LAR, as supplemented by letters dated June 24, and August 31, 2026, describes how the licensee considered the heating created by gamma radiation produced by the decay of Co-60 and the licensee's conclusion that the decay heat would remain within the safety analyses' values. The licensee assumed all COBA generated heat is deposited in the core as an additional component of decay heat and remains constant for all safety analyses because of relatively long cobalt-60 half life. The licensee also calculated the additional heat added to the core by Co-60 production.

The licensee stated that the total decay heat for each core will be ensured to remain below the calculated conservative total value by calculating the total Co-60 curie production in a given cycle and comparing it to the calculated total decay heat from COBA that is included in safety analysis evaluations.

The NRC staff finds that because the licensee's assessment of COBA decay heat was conservatively determined and that the licensee will ensure that calculated decay heat for each core is bounded by the value used in the safety analysis, the NRC staff has reasonable assurance the Co-60 implementation conforms to the chapter 15 analyses in the UFSAR and, therefore, is acceptable.

3.6.3 Evaluation of Non-LOCA Transients

Section 8.3, "Non-LOCA Transients," of attachments 2 and 3 to the LAR, as supplemented by letters dated June 24, and August 31, 2026, describes the licensee's assessment of the effects of Co-60 production on the non-LOCA analyses described in chapter 15 of the UFSAR. The licensee's analysis concluded that there was no impact on the non-LOCA analyses from the Co-60 production because there would be no adverse effects on the control rod drop time, reactor trip rod positions versus time curve, nor on the core bypass flow fraction and core pressure drop. The licensee's analysis concluded that there would not be significant impacts from Co-60 production on rod reactivity insertion versus position curve nor on the rod cluster control assembly (RCCA)-related reload safety analysis checklist (RSAC) parameters, including the core axial power shapes, peaking factors, and axial offset limits. The licensee's analysis also concluded that there would be no adverse effect from Co-60 production on the decay heat generated in the core following a reactor trip. The non-LOCA analyses at Salem use ANSI/ANS-5.1-1979 decay heat plus two standard deviations (2σ), and the licensee concluded this existing margin is sufficient to offset the additional decay heat associated with activation of cobalt slugs. Overall, the licensee concluded that implementation of Co-60 production does not adversely affect non-LOCA transient inputs, analytical assumptions, or acceptance criteria. The licensee indicated that all relevant safety analysis limits remain valid for cores containing COBA assemblies and that COBA rodlets maintain structural and thermal integrity under analyzed non-LOCA event conditions.

The NRC staff evaluated the licensee's proposed use of COBA assemblies and their effects on the UFSAR, chapter 15 non-LOCA transient safety analyses described in the section 8.3 of attachments 2 and 3 of the LAR; the responses to audit items 15, 19, 20, and 21 in the supplement dated June 24, 2026; and the response to audit item 25 in the supplement dated August 31, 2026. The NRC reviewed the licensee's core physics inputs, thermal-hydraulic parameters, analytical limits, component performance under anticipated operational

occurrences, and decay heat model application. The NRC staff focused its review on the scope of changes introduced by the COBA assemblies and verified that analytical methodologies, assumptions, and licensing-basis requirements used in the non-LOCA analyses remain applicable. The NRC staff evaluated key RSAC parameters, thermal-hydraulic inputs (such as core bypass fraction and pressure drop), departure from nucleate boiling ratio (DNBR) performance, COBA component behavior under the RCCA ejection event, and decay heat methodology applicability. The NRC staff assessed whether the inclusion of COBA assemblies would affect key analytical inputs or system responses to determine whether the licensee's evaluations demonstrate continued conformance with the conservative assumptions in the analyses in chapter 15 of the UFSAR and AEC proposed GDC 6, as described in the UFSAR, related to fuel integrity during anticipated operational occurrences.

3.6.3.1 Evaluation of Impact of Cobalt Burnable Absorbers on Core Power Distributions

The NRC staff reviewed the licensee's assessment of COBA-related effects on core power distributions, including axial power shapes, peaking factors, axial offset limits, and RCCA reactivity-insert characteristics. The licensee stated that the introduction of COBA assemblies does not produce significant changes in the key RSAC parameters, including core power distributions, and that any observed differences between COBA and non-COBA reload patterns fall within the typical range of designer variability seen during normal reload development.

The NRC staff reviewed the information in the "Audit Item 15" section of the supplement dated June 24, 2026, regarding numerical comparisons, confirmation of acceptance criteria, identification of any localized peaking effects, and clarification of conditions under which COBA effects might be nonminimal. The NRC staff also reviewed axial power shape plots, comparative data, and supporting analyses discussed in supplement. The NRC staff determined that these analyses demonstrate that COBA inserts are explicitly modeled within the NRC-approved core reload methodology in WCAP-9272-P-A (Reference 13). The NRC staff confirmed that

[[

]] The NRC staff determined that the limits on peaking factors, axial offset, and axial power distribution in the TS 3/4.2 LCOs would continue representing the lowest functional capability or performance levels required for safe operation per 10 CFR 50.36(c)(2), and that the licensee's evaluation provides reasonable assurance that COBA-related effects on core power distributions are adequately bounded by existing methodologies and acceptance criteria.

Therefore, the NRC staff determined that the licensee adequately demonstrated that key RSAC parameters, including axial power shapes, peaking factors, and axial offset limits, would remain within acceptable bounds following the implementation of COBA assemblies. The NRC staff determined that any cycle-specific variations are minor and would be effectively managed through the established reload safety analysis process, thus providing reasonable assurance of continued conformance with AEC proposed GDC 6, as described in the UFSAR, and maintaining core safety margins.

3.6.3.2 Evaluation of Impact of COBA Assemblies on Thermal-Hydraulic Inputs: Bypass Flow Fraction

The NRC staff reviewed the licensee's assessment of the hydraulic impacts associated with installing COBA assemblies. The licensee stated that the inclusion of COBA assemblies would have no adverse effect on the core bypass flow fraction and core pressure drop. The NRC staff audited the bypass flow calculation confirming that core bypass flow fraction and core pressure drop remain within the design and analysis of record limits with COBA assemblies in the planned locations. In its supplement dated June 24, 2026, pertaining to audit item 19, the licensee indicated that the subject calculation used the Thermal Hydraulic Reactor Internal Vessel Evaluation (THRIVE) code to model reactor-vessel flow distribution and identify all bypass flow paths. Because COBA rods occupy previously open thimble tubes, the licensee performed a simplified hydraulic network calculation to quantify thimble tube bypass flow and then integrated this contribution into the overall bypass flow distribution calculated by THRIVE. The resulting calculated bypass flow was confirmed to be within the core bypass design flow used in the AOR supporting the current licensing basis (CLB).

Based on its review of the licensee's THRIVE calculational results, the NRC staff determined that the licensee adequately demonstrated that installation of COBA rodlets [] in a manner that affects the UFSAR, chapter 15 non-LOCA analyses.

3.6.3.3 Evaluation of Impact of COBA Assemblies on DNBR Performance

The NRC staff reviewed the licensee's assessment of the effects of COBA assemblies on DNBR performance for anticipated operational occurrences. The NRC staff audited the licensee's methodology and results for demonstrating that DNBR margins for anticipated operational occurrences are not reduced by the presence of COBA assemblies, addressing [] COBA rods. As indicated in its supplement dated June 24, 2026, regarding audit item 20, the licensee applied the NRC-approved WCAP-9272-P-A reload methodology, demonstrating that COBA-related changes to core power distributions [] and consistent with normal cycle-to-cycle variations.

The licensee's thermal-hydraulic analyses indicated that the minimum DNBR calculated in the AOR supporting the CLB remains bounding for cores containing COBA rodlets. The licensee's analyses were performed using: the NRC-approved WRB-2M DNBR correlation in WCAP-15025-P-A, Revision 0 (Reference 17); VIPRE-W subchannel methods in WCAP-14565-P-A, Revision 0 (Reference 16); and the existing 95/95 DNBR statistical framework in WCAP-18240-P-A, Revision 0 (Reference 18). The licensee also confirmed that insertion of COBA rodlets into guide thimble tubes []

[]

Based on its review of this information, the NRC staff determined that DNBR margins remain preserved and that the approved DNBR methodology remains applicable without modification for cores containing COBA assemblies.

3.6.3.4 Evaluation of Impact of COBA Assemblies on RCCA Ejection Event

The NRC staff reviewed the licensee's assessment of the thermal response of COBA assemblies during the RCCA ejection event described in section 15.4.7 of the UFSAR. The

NRC staff audited information supporting that COBA components maintain structural and thermal integrity under the same conditions. In its supplement dated June 24, 2026, regarding audit item 21, the licensee indicated that a supplemental thermal analysis was performed using TWINKLE and FACTRAN, consistent with Salem's CLB methodology, as discussed in chapter 15 of the UFSAR. In this supplemental analysis, [[

]]

The NRC staff determined that the supplemental thermal analysis used conservative assumptions and the TWINKLE and FACTRAN methodologies used in the AOR, as described in chapter 15 of the UFSAR, and that the results bounded the expected thermal response, confirming that COBA components maintain structural and thermal integrity during an RCCA ejection event. Therefore, the NRC staff determined that the licensee's conservative analysis of the RCCA ejection event adequately demonstrates that cobalt slug and capsule temperatures would remain below melting limits with adequate margin and, thus, maintain structural and thermal integrity of COBA components under the limiting rod ejection scenario.

3.6.3.5 Evaluation of Decay Heat Verification for Non-LOCA Analyses

The NRC staff reviewed the licensee's assessment of decay heat assumptions used in the non-LOCA transient analyses. The NRC staff audited how the licensee would verify that decay heat predictions, including contributions from COBA assemblies, would remain conservative for applicable safety analyses. In its supplement dated August 31, 2026, regarding audit item 25, the licensee indicated that the non-LOCA analyses rely on [[

]] and do not use cycle-specific decay heat inputs. Because these analyses are not sensitive to small variations in isotopic inventories, the NRC staff determined that COBA-related decay heat does not affect initial conditions or system responses modeled in the non-LOCA transients.

The NRC staff reviewed and confirmed that COBA-induced isotopic effects are like those associated with existing burnable absorber behavior showing that the burnable absorber-induced isotopic effects are second order. The NRC staff also reviewed the licensee's configuration-control and reload-evaluation processes described in its supplement dated August 31, 2026, and found that these processes provide reasonable assurance that any future reload containing COBA assemblies will continue to meet applicable decay heat assumptions. The NRC staff determined that COBA activation does not alter the conservatism inherent in the [[
]] Therefore, the NRC staff determined that the licensee adequately demonstrated that use of [[
]] continues to provide appropriate margin for non-LOCA analyses.

3.6.3.6 Evaluation of Scope of Non-LOCA Transients Reviewed

The NRC staff confirmed that the licensee's assessment addressed the full set of non-LOCA events described in chapter 15 of the UFSAR, as listed in table 8.3-1 of attachment 3 to the

LAR. The NRC staff's technical evaluation in sections 3.6.3.1 through 3.6.3.5 of this safety evaluation verified that key RSAC parameters, analytical limits, DNBR margins, decay-heat assumptions, and component performance remain valid and conservative for cores containing COBA assemblies.

Except for the RCCA ejection event, the licensee's non-LOCA analyses showed significant margin to material melting limits, demonstrating that COBA rod and capsule integrity is maintained. For the RCCA ejection event, the licensee performed a supplemental evaluation that bounded expected plant conditions and demonstrated that cobalt slug temperatures remain below the melting limit with adequate margin. Based on its review of this information, the NRC staff determined that both the standard non-LOCA analyses and the supplemental RCCA ejection evaluation are adequate to support implementation of COBA assemblies.

3.6.3.7 Conclusion

Based on the technical evaluation above, the NRC staff determined that the inclusion of COBA assemblies in the reactor core does not adversely affect the non-LOCA safety analyses or associated analytical limits. The licensee has demonstrated that key RSAC parameters, including axial power shapes, peaking factors, and axial-offset limits, remain conservative with the inclusion of COBA assemblies. Therefore, the NRC staff concludes that the non-LOCA analysis in the LAR, as supplemented, continues to conform to chapter 15 of the UFSAR regarding the use of conservative assumptions to maintain adequate margin to fuel safety limits. The licensee has also demonstrated that the addition of COBA assemblies does not adversely affect the non-LOCA transient safety analysis or associated safety analytical limits. Because the analytical outcomes remain consistent with established fuel integrity criteria, the NRC staff finds it has reasonable assurance that the proposed changes conform to AEC Proposed GDC 6, as described in the UFSAR. Therefore, the NRC staff concludes that the proposed use of COBA assemblies is acceptable with respect to non-LOCA transient analyses.

3.6.4 Evaluation of Loss of RHR at Midloop/Natural Circulation Cooldown

Section 8.4, "Loss of RHR at Midloop / Natural Circulation Cooldown," of attachments 2 and 3 to the LAR, as supplemented by letter dated August 31, 2026, describes the licensee's assessment of the increase in decay heat from the COBAs on the loss of residual heat removal (RHR) at the middle of the hot leg (midloop) and natural circulation cooldown (NCC) analyses. The licensee stated in its LAR that the implementation of COBAs will result in an increase in the decay heat produced by the reactor core, which in turn results in a shorter time for the core to heat up to saturation and boiling conditions. The licensee stated that the Electric Power Research Institute (EPRI) developed a set of standard values for decay heat as a function of time after shutdown using the ANSI/ANS-5.1-1979 methodology with conservative inputs. The licensee stated that these standard decay heat values were published in the report, NSAC-176L, "Safety Assessment of PWR Risk During Shutdown Operations," EPRI Outage Risk Assessment and Management (ORAM) Program (prepared by Westinghouse Electric Corporation), Final Report," August 1992. The licensee concluded that the current loss of RHR at midloop analysis and NCC analysis have sufficient margin to account for the additional decay heat from COBAs and, therefore, that there is no impact to the loss of RHR at midloop or to NCC. As such, the licensee also determined that there is no impact to the licensing basis for those functional areas.

The NRC staff audited the licensee's analyses of the decay heat associated with the COBAs. In its supplement dated August 31, 2026, regarding audit items 6, 25, and 36, the licensee described the value of added decay heat and how it was considered in its analyses. The NRC staff finds that the licensee using the TREAT software to assess the existing NCC AOR decay heat with the small fraction of added decay heat from Co-60 production is acceptable because of the minor differences between the decay heat values used in the AOR and the Westinghouse-evaluated bounding decay heat for the loss of RHR at midloop event. The NRC staff reviewed the information in table 1, "Comparison of Reference 5 and NSAC-176L Decay Heat Values," and table 2, "Comparison of Reference 7 and NSAC-176L Decay Heat Values," under the "Audit Item 36" section of the supplement dated August 31, 2026. The NRC staff confirmed that both the AOR values of decay heat and Westinghouse conservative values have minor differences with the EPRI NASC-176L decay heat values shown in the supplement. The NRC staff determined that given the conservative nature of the NSAC-176L decay heat values with a small fraction of added decay heat caused by Co-60 production, the current AORs have sufficient conservatism to account for the decay heat from Co-60 production.

The NRC staff determined that the information in tables 1 and 2 in the "Audit Item 36" section of the supplement dated August 31, 2026, demonstrates adequate margin in the NCC and RHR at midloop analyses. Therefore, the NRC staff finds that that it has reasonable assurance that the NCC and RHR at midloop analyses have sufficient margin to account for the additional decay heat from COBAs and, therefore, that there is no impact to the loss of RHR at midloop or to NCC.

3.6.5 Evaluation of FLEX Operation, Shutdown Cooling, and SFP Cooling and Boron Dilution

In section 8.5, "FLEX Operation, Shutdown Cooling, and SFP Cooling and Boron Dilution," of attachments 2 and 3 to the LAR, as supplemented by letter dated August 31, 2026, the licensee described its assessment for determining how the added decay heat from the COBAs would impact the flexible coping strategies (FLEX) analyses that address an extended loss of alternating current power (ELAP), shutdown cooling (RHR cooldown), SFP cooling and heat up, and SFP boron dilution analyses that are input to the SFP criticality safety analysis (CSA).

Section 8.5 of attachment 3 to the LAR lists the screening criteria that the licensee established based on a generic impact evaluation, for determining if the results of the following FLEX-related analyses bound the use of COBAs:

- (a) containment pressure and temperature response
- (b) RCS makeup flow
- (c) auxiliary feedwater makeup flow
- (d) condensate inventory

In the "Audit Item 38" section of the supplement dated August 31, 2026, the licensee indicated that for item (a), it used ANSI/ANS-5.1-1979 $\sigma = 0$ consistent with the usage of the EPRI FLEX methodology, Modular Accident Analysis Program (MAAP) 4.0.6. For items (b), (c), and (d), the licensee stated that it used ANSI/ANS-5.1-1979 $\sigma = 2$. For all four calculations, the licensee determined that the bounding contribution of **[[]]** from COBA fuel inserts is not significant for these beyond-design-basis (BDB) analyses given the duration of the event.

The licensee stated that best estimate and engineering judgment are permitted for beyond design basis analyses.

In section 8.5 of attachment 3 to the LAR, the licensee stated that per section 5.5.7.3.4 of the UFSAR, the safe shutdown design basis is hot standby (Mode 3). The licensee stated that single-train RHR cooldown represents the limiting analysis relative to added COBA decay heat, and that the analysis results are conservative because of the use of the BOP-FR-8 model and relevant cooldown times that are less than 72 hours.

For bringing a unit to Mode 3 using the limiting single train of RHR cooldown, including COBA decay heat, the licensee's evaluation provided conservative results from using the BOP-FR-8 model and relevant cooldown times less than 72 hours. The licensee stated that its sensitivity study determined that the impact of COBA decay heat results in an increase of about 2 hours for the analyzed single-train cooldown cases. The licensee concluded that because it has been shown that the plant can be cooled to cold shutdown (Mode 5) by manual actions with existing equipment, the addition of COBA is acceptable but would prolong the time to reach a specific RCS Mode 5 temperature.

The licensee indicated that there would be no effect on the FLEX mitigative actions with the addition of COBAs. The licensee indicated that while single-train RHR cooldown times would be longer, the RCS can be cooled to 200 °F (Mode 5 entry) over a period that would be extended by approximately 4 to 5 percent because of the presence of COBA. The licensee concluded that all design and licensing requirements for FLEX operation, RHR cooldown, and SFP cooling and boron dilution will continue to be met.

Section 3.7.4 of this safety evaluation addresses the licensee's SFP cooling assessment. Section 3.7.2 of this safety evaluation addresses the SFP time-to-boil assessment. The NRC's safety evaluation for Amendment Nos. 354 and 336 (Reference 7) addresses the licensee's SFP boron dilution assessment.

Regarding the licensee's use of the ANSI/ANS-5.1-1979 model for all FLEX analyses; the NRC staff finds that the licensee's FLEX assessment acceptable because the licensee demonstrated that even though the COBA decay heat increased the single-train RHR cooldown time by 2 hours, this increase remains within the acceptable overall cooldown time of less than 72 hours. Therefore, the NRC staff determined that the results of the assessments for FLEX and RHR cooldown demonstrated there is no substantive effect on the FLEX mitigative actions with the addition of COBA inserts.

3.6.6 Evaluation of Loss-of-Coolant Accident

Section 8.6, "Loss of Coolant Accident," of attachments 2 and 3 to the LAR, as supplemented by letters dated June 24, and August 31, 2026, describe the licensee's assessment of impacts of the proposed insertion of COBAs on its LOCA analysis. The licensee's analysis is divided into four parts:

- COBA capsule survivability during a LOCA
- impacts of the COBA on the small-break LOCA (SBLOCA) and large-break LOCA (LBLOCA) analysis

- impacts of COBA insertion on the containment integrity analysis for LOCA
- impacts of COBA insertion on the post-LOCA long-term cooling analysis

The licensee's analysis encompasses the impacts of the LOCA-induced heat up on COBA integrity and the potential impacts of the additional heat generated by the COBAs on the analysis of existing figures of merit for the LOCA event associated with core coolability and containment integrity. The NRC staff's evaluation of these topics is in the following four subsections of this safety evaluation.

3.6.6.1 Evaluation of COBA Capsule Survivability

Section 8.6.1, "COBA Capsule Survivability," of the licensee's submittal discusses survivability of COBA capsules under postulated LOCA conditions. The licensee's use of the term "survivability" in section 8.6.1 of attachment 2 connotes that the leak-tight, **[]** COBA capsule credited with retaining the radioactive source material maintains its integrity, thereby ensuring that the calculated radioactive source term for design-basis accidents is not exceeded. The licensee's assessment of survivability under postulated LOCA conditions considered expected failure modes including the potential for capsule rupture, melting of COBA components, and eutectic formation.

Capsule Rupture

In section 8.6.1 of attachment 2 to the LAR, regarding capsule rupture, the licensee described three experimental programs relevant to assessing the bursting of steel pressure tubes at elevated temperature. The licensee concluded that the internal pressure necessary to rupture COBA capsules would be 520 pounds per square inch absolute (psia), which the licensee reduced to 494 psia to account conservatively for anticipated variation in dimensional tolerances. The licensee further stated that a conservatively calculated capsule internal pressure during a postulated LOCA would not exceed 334 psia, which is indicative of significant margin to rupture.

The licensee's analysis of the test data found that, in light of the COBA dimensions in table 8-1, "Relevant burst testing measurements," of attachment 2 to the LAR and the experimentally determined hoop stress values in table 8-3, "Burst Temperature vs Pressure data from literature," of the same attachment, the rupture pressure for COBA capsules would be approximately 387 psia, with an estimated rupture time of approximately 101 seconds. The licensee stated that the assessment of this testing further demonstrates that no rupture would be expected considering the maximum capsule internal pressure expected during a postulated LOCA is 334 psia. The licensee also stated that, prior to burst, the COBA capsules would be expected to experience diametral strain that would either reduce their internal pressure or result in contact with the inner wall of the COBA rodlet, either of which would reduce the stress across the COBA capsule wall and consequently reduce the potential for capsule rupture.

The NRC staff reviewed the licensee's assessment of COBA capsule rupture and found that the licensee had (1) used appropriate, material-specific data for assessing the potential for capsule rupture, (2) used applicable equations for determining capsule internal pressures and hoop stresses, and (3) incorporated sufficient design margins to provide reasonable assurance that COBA capsule rupture would not occur during a postulated design-basis LOCA.

Melting of COBA Components

In section 8.6.1 of attachment 2 to the LAR, the licensee indicated that upper bound temperature conditions from its safety analysis events, including the postulated LOCA, remain below the melting temperatures of both [] The NRC staff notes that published data for both alloys indicates that their melting temperatures exceed 2,500 °F, which is greater than the allowable peak cladding temperature specified in 10 CFR 50.46 for zirconium fuel rod cladding. The published melting point of cobalt is even higher (i.e., in excess of 2,700 °F). In light of the high melting temperatures for COBA constituent components, and further considering that the hottest temperatures in the reactor core are expected to exist at the fuel rods (i.e., which are the predominant source of heat generation in the core) the NRC staff finds that it has reasonable assurance that the COBA capsules would not exceed their melting temperature for the existing plant configuration.

Eutectic Formation

In section 8.6.1 of attachment 2 to the LAR, the licensee indicated that in addition to considering the melting of the individual solid materials comprising the COBAs, it considered the potential for eutectic formation to result in component melting at a temperature below that of the individual constituent materials. []

[]

The licensee assessed the potential for such eutectic formation, stating that, in contrast to normal operation, during a postulated LOCA, hard contact may occur within the COBA between the [] Under this condition, the licensee stated that []

[] The licensee indicated that the eutectic reaction kinetics are such that (1) the time duration required to damage COBAs exceeds the time at peak temperature during a postulated LOCA and (2) the reaction rate falls precipitously as temperature is reduced (e.g., a temperature reduction of 100 °C from the peak temperature would reduce the reaction rate by a factor of 60).

In the “Audit Item 16” section of its supplement dated August 31, 2026, the licensee indicated that protective oxides are expected to form on both the outer tube and the capsules, which may limit eutectic formation by restricting the potential for hard contact between the two materials. The licensee further indicated that the eutectic formation process would result in localized melting (i.e., at the material interface) rather than bulk melting throughout the component. The licensee further stated in the supplement that []

[]

In the “Audit Item 16” section of its supplement dated August 31, 2026, the licensee indicated that []

[] The licensee also discussed plant-specific information concerning the analytical results for the

LOCA, noting that in light of the peak cladding temperature of 2,129 °F and a total heat up duration of 600 seconds, the time duration for which a temperature of [[]] would be exceeded would be approximately [[]] which is much less than the [[]]

The NRC staff determined that the licensee's assessment of [[]] eutectic formation generally consistent with the expected physical behavior for these materials. While the licensee did not expect eutectic formation to impact COBA survivability, the NRC staff found it reasonable to expect that [[]]

[[]] The NRC staff also determined two key conservatisms associated with the temperature conditions considered in the licensee's analysis of eutectic formation for the COBAs:

- The licensee's assessment of eutectic formation considered peak cladding temperature conditions for a limiting fuel rod. However, local temperatures experienced by COBAs during a LOCA are expected to be significantly lower than those for fuel rods, particularly the hot rod that is subject to bounding peaking factors and other conservative assumptions.
- The licensee's LOCA analysis is performed with a conservative evaluation model that conforms to the required and acceptable features delineated in Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, which adds significant conservatism to the calculated values for the peak cladding temperature and other figures of merit for the LOCA.

Based upon its evaluation above, the NRC staff has reasonable assurance that eutectic formation under LOCA conditions will not adversely affect the survivability of COBA capsules.

3.6.6.2 Evaluation of Small Break and Large Break LOCA

In section 8.6.2, "Small Break and Large Break LOCA," of attachments 2 and 3 to the LAR, the licensee discussed the potential for impacts of COBA insertion on the results of the LOCA analysis performed to demonstrate compliance with the acceptance criteria specified in 10 CFR 50.46.

In attachment 2 to the LAR, the licensee stated that its review of expected impacts on the RSAC (key safety parameters) of including COBA in the core design, the power shape distributions assumed in the LOCA analyses, and the peaking factor versus burnup behavior are not expected to be impacted by the inclusion of COBA. The licensee also stated that its review of the impacts on the fuel generalized energy deposition model concluded the overall gamma energy deposition in the fuel remains appropriate.

In attachment 3 to the LAR, the licensee further discussed considerations associated with radioactive decay heat and indicated that the fraction of the total core decay heat generated by COBAs is quite small and that this decay heat is generated outside of the fuel rod and, thus, does not directly impact cladding temperature. The licensee stated that the additional energy produced in the core can impact the local fluid conditions and, thus, the ability to transfer heat from the fuel rods. The licensee stated that the amount of decay heat produced by the COBAs is significantly lower than the decay heat produced by the fuel rods and would have negligible effect on the thermal-hydraulic response calculations contained in the SBLOCA and LBLOCA

analyses. The licensee further indicated that the decay heat modeling in its analyses for both LBLOCAs and SBLOCAs follows the conservative approach prescribed in Appendix K to 10 CFR Part 50, which includes a 20 percent uncertainty, among other conservatisms.

In attachment 3 to the LAR, the licensee concluded that coolability of both the COBA rodlets and core will be maintained and that the COBA assemblies will have a negligible impact on the LBLOCA and SBLOCA analyses, leading to a 0 °F peak cladding temperature (PCT) impact for 10 CFR 50.46 reporting purposes.

In section 8.6.2, "Small Break and Large Break LOCA," of the "Attachment 1: Decay Heat Summaries" section of its supplement dated August 31, 2026, the licensee described its analyses for the LBLOCA and SBLOCA that simulated the heat addition of the COBAs by commensurately increasing **[[** **]]** This modeling approach conservatively did not credit the heat absorption capacity of the COBA assembly or guide thimble. Using this approach, the licensee determined that the localized heating associated with the COBA rodlets would result in a 12 °F increase in LBLOCA PCT, and a 5 °F increase in PCT for the SBLOCA. The licensee indicated that these predicted impacts from the planned insertion of COBAs would be used for reporting purposes pursuant to the requirements specified in 10 CFR 50.46.

The NRC staff found that the licensee's assessment of the impacts of COBA insertion on the LBLOCA and SBLOCA analysis for Salem was performed in an appropriate manner that estimated impacts using the approved evaluation model for Salem with a conservative approach for incorporating the COBA decay heat contribution. Therefore, the NRC staff is reasonably assured that the licensee (1) has acceptably estimated the impact of COBA insertion on its analysis of the LOCA pursuant to 10 CFR 50.46(a)(3)(i) and (2) maintains compliance with applicable acceptance criteria specified in 10 CFR 50.46(b).

3.6.6.3 Evaluation of LOCA Containment Integrity

Section 8.6.3, "LOCA Containment Integrity," of attachments 2 and 3 of the LAR, as supplemented by letters dated June 24, and August 31, 2026, describes the licensee's analysis of the effect of COBA assemblies in the core on the long-term LOCA containment integrity analysis, including overall peak containment pressure and temperature, and the long-term containment temperature response used to evaluate equipment qualification (EQ).

In section 8.6.3 of attachment 3 to the LAR, the licensee concluded the following:

- The current analyses support plant operation in the reactor vessel upflow configuration as described in WCAP-18707-P, Revision 0, "Reactor Internals Upflow Conversion Program Engineering Report for Salem Nuclear Generating Station Unit 1," May 2022, and WCAP-18659-P, Revision 0, "Reactor Internals Upflow Conversion Program Engineering Report for Salem Nuclear Generating Station Unit 2," August 2021.
- The long-term LOCA mass and energy (M&E) releases and containment response analyses that are documented in section 9.4 of WCAP-18659-P bound both units.
- The inclusion of COBA rod assemblies will not increase the RCS initial temperatures, pressure, primary side fluid volume, secondary side fluid volume, secondary side metal mass, or the initial core stored energy.

- There would also be only a negligible increase in metal mass in the reactor vessel.
- There would be a slight increase in the decay heat energy release rate. The peak containment pressure and temperature results remain unchanged from those reported in table 9-26 of WCAP-18659-P. Although additional energy will result in a slight increase in the containment pressure and steam temperature at 24 hours after accident initiation and an increase in the long-term containment temperature results used in the EQ program, the transient figures provided in section 9.4.2.5 of WCAP-18659-P remain valid.

The NRC staff reviewed the following aspects of the licensee's LOCA containment integrity analysis:

- containment M&E release analysis
- short-term and long-term containment pressure and temperature response
- impact on containment integrated leak rate test (ILRT) pressure
- impact on containment structural design temperature
- impact on containment EQ profile
- containment sump temperature response
- net positive suction head (NPSH) margin analysis of pumps that draw water from the sump during the LOCA recirculation phase
- comparison of containment accident pressure used in calculating NPSH margin to the saturation pressure at sump temperature

The NRC staff audited the licensee's analyses supporting the conclusions in the LAR. In the "Audit Item 4 and 5" section of the supplement dated June 24, 2026, the licensee confirmed that its long-term LOCA M&E release calculations addressed the issues identified in the Westinghouse nuclear safety advisory letters (NSALs) 06-06 (Reference 30), 11-5 (Reference 31), and 14-2 (Reference 32) regarding the Westinghouse LOCA M&E release methodology in report WCAP-10325-P-A. The NRC staff determined that because the licensee's LOCA M&E release AOR considered the impact of these NSALs, the only impact on M&E release would be from the COBA decay heat. Because there will be an increase to the decay heat energy release rate contribution from the COBAs that would be released during a LOCA, the NRC staff reviewed the licensee's analysis of its impact to the M&E release AOR. The NRC staff reviewed the information in table 8.6.3-1, "Salem Units 1 and 2 LOCA Containment Response Results (Loss-of-Offsite Power Assumed)," in attachment 3 to the LAR and the table of NPSH analyses results included on page 8 of enclosure 1 to the supplement dated June 24, 2026.

The NRC staff made the following technical determinations about the impacts from the COBA decay heat based on its review of the information section 8.3.6 of attachments 2 and 3 to the LAR, as supplemented by letter dated June 24:

- The M&E release would increase by ~~[[~~ ~~]]~~¹
- The short-term containment response would not be affected.
- The long-term containment pressure and temperature would increase slightly but remain bounded by the design values.
- The current containment ILRT pressure would not be affected.
- The sump water temperature increase over the AOR values is insignificant.
- The containment wall temperature would not be affected and would remain bounded by its design temperature.
- The containment EQ profile would remain bounding.
- The limiting NPSH margin for the RHR pumps during the LOCA recirculation phase would remain conservatively positive without crediting any containment accident pressure above the sump water saturation pressure.

The NRC staff finds it acceptable that the addition of COBAs does not affect the short-term containment pressure and temperature responses because it depends on metal sensible heat and the RCS initial pressure and temperature which remain unchanged from the AOR. The NRC staff also finds the licensee's long-term containment pressure and temperature response analysis with COBA acceptable based on the following:

- The LOCA M&E release analysis is performed using the NRC-approved methodology described in WCAP-10325-P-A (Reference 14) and addresses the errors reported in the three NSALs referenced above.
- The containment pressure and temperature transient analysis is performed using the COCO computer code described in WCAP-8327, "Containment Pressure Analysis Code (COCO)," July 1974, which is also credited in section 15.4.8.1.1 of the UFSAR.
- The containment peak pressure and peak vapor temperature remain bounded by their design values.
- The long-term pressure and temperature transients show an insignificant increase from the AOR transient results.
- The containment structural design temperature is not affected because the containment peak vapor temperature is unchanged from its AOR value.

¹ In its supplement dated August 31, 2026, the licensee discussed an updated bounding COBA decay heat value of ~~[[~~ ~~]]~~ and indicated that some analyses use this updated value. The licensee clarified during the audit that its LOCA M&E releases and containment response analyses used a value of ~~[[~~ ~~]]~~, as discussed in its supplement dated June 24, 2026, which the licensee indicated was also conservative.

Based on the above results, NRC staff finds that adequate margin exists for single RHR pump operation and two RHR pump operation for each mode of recirculation operation. In addition, the NRC staff determined that the sump strainer design eliminates the risk of damaging the pumps via flashing or vortexing, which is consistent with Generic Letter 2004-02 (Reference 19). Therefore, the NRC staff finds that the proposed changes conform to the following:

- AEC proposed GDC 10 and 42, as described in the UFSAR, because the capability of each component and system to perform its required function is not impaired and containment integrity is maintained by the effects of a large break LOCA
- AEC proposed GDC 41, as described in the UFSAR, because the containment heat removal system will maintain its long-term cooling function by removing additional decay heat released from the COBAs assuming a failure of a single active component
- AEC proposed GDC 49, as described in the UFSAR, because the containment and its associated heat removal systems can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA

3.6.6.4 Evaluation of Post-LOCA Long-Term Cooling

In section 8.6.4, "Post-LOCA Long Term Cooling," of attachments 2 and 3 to the LAR, the licensee described its assessment of the potential for impacts of COBA insertion on the results of the post-LOCA long-term cooling analysis. In its supplement dated August 31, 2026, the licensee corrected information in section 8.2 of attachment 2 to the LAR concerning the bounding decay heat value for the proposed COBA insertion. The licensee stated that the corrected decay heat power of $[[\quad \quad \quad]]$ was determined by considering the maximum number of COBAs inserted into the core with each COBA at its maximum achievable decay heat power. The licensee further stated in its supplement dated August 31, 2026, that its assessment of long-term cooling considered this decay heat power.

In attachment 3 to the LAR, as supplemented by letter dated August 31, 2026, the licensee addressed the following elements of its post-LOCA long-term cooling analysis: maintaining subcriticality, boric acid precipitation control, and decay heat removal.

Maintaining Subcriticality

With respect to maintaining subcriticality, the licensee stated in attachment 3 to the LAR that post-LOCA subcriticality calculations determine a mixed mean boron concentration in the sump, and that the calculations do not consider decay heat. The licensee's statement that decay heat is not considered in the calculation indicates that no credit is taken for any temperature increase that the decay heat from COBA insertion could provide. The licensee concluded that post-LOCA subcriticality is not impacted by the COBA inserts.

The NRC staff finds this approach reasonable because mitigation of a postulated LOCA with very low COBA decay heat loads could be necessary. Furthermore, while the addition of cobalt and other constituent materials comprising the COBAs could slightly increase parasitic neutron absorption, no credit for such absorption is warranted as the reactor core design would be expected to adjust for even these slight impacts. Therefore, the NRC staff is reasonably assured

that the introduction of COBAs would not adversely affect the licensee's assessment that post-LOCA subcriticality is maintained.

Boric Acid Precipitation Control

With respect to boric acid precipitation control, the licensee indicated in attachment 3 to its LAR that the analysis is performed to determine the appropriate time to switch to hot leg recirculation to avoid the precipitation of boric acid in the core. The licensee stated that its CLB switchover times are conservatively reported as 14 hours for Unit 1 and 6.5 hours for Unit 2 after LOCA initiation.

As described above in section 3.6.6.2 of this safety evaluation, the licensee has indicated that the COBAs produce a small fraction of the total core decay heat power. In attachment 3 to the LAR, the licensee stated that as time goes on, the relatively small amount of heat generated by the COBA inserts becomes more significant relative to the decay heat produced by the fuel rods. The licensee also stated that the decay heat from COBAs increases to just over 1 percent of the Appendix K decay heat by 14 hours after the LOCA. In its supplement dated August 31, 2026, the licensee stated that the time reported in the UFSAR to switch to hot leg recirculation is conservatively rounded down from the calculated time and that there is sufficient margin between the calculated and reported hot leg switchover times to accommodate the additional decay heat from the COBA assemblies.

The NRC staff finds that it has reasonable assurance that adequate boric acid precipitation control is maintained with introduction of COBAs because the licensee has verified that the additional decay heat from the COBAs is within the conservative margin for determining the CLB hot leg switchover times.

Decay Heat Removal

With respect to decay heat removal, the licensee stated in attachment 3 to the LAR that its calculations have verified that post-LOCA minimum flow requirements for ensuring sufficient vessel inventory during cold-leg and hot-leg recirculation are satisfied. The licensee indicated that the incremental change in decay heat associated with the COBAs does not impact its ability to continue to satisfy minimum coolant flow requirements. In its supplement dated August 31, 2026, the licensee indicated that additional conservative margins are available to offset the additional decay heat from the COBAs, but they are not quantified.

The NRC staff determined that it has reasonable assurance that adequate decay heat removal is maintained with introduction of COBAs because the licensee has verified that the analyzed minimum flow rates establish sufficient margin to account for the slight increase in decay resulting from the insertion of the COBAs. The NRC staff determined that the additional decay heat contribution from the COBAs is incremental and, based upon the licensee's verification that minimum flow rates continue to be satisfied following the insertion of COBAs, has reasonable assurance that the conservative margins in the licensee's analysis are sufficient to accommodate the additional impact.

3.6.7 Evaluation of Steam Generator Tube Rupture

Section 8.7, "Steam Generator Tube Rupture," of attachment 3 to the LAR, as supplemented by letter dated August 31, 2026, describes the licensee's assessment of the impact of the decay

heat added by the COBA assemblies on the steam generator tube rupture (SGTR) analysis. The licensee concluded that the decay heat added by the COBA assemblies would not change the analysis decay heat modeling or the RHR system cut-in time, is not significant for the 32-hour duration of the SGTR event, and would not change the analysis modeling. The NRC staff reviewed the licensee's assessment and determined that it provides reasonable assurance that the decay heat added by the COBA assemblies would not significantly affect the SGTR analysis.

3.6.8 Evaluation of Steamline Break M&E and Steam Release for Dose

In section 8.8, "Steamline Break M&E and Steam Release for Dose," of attachment 3 to the LAR, as supplemented by letter dated August 31, 2026, the licensee described its assessment of the impacts of a maximum decay heat contribution from Co-60 production on the steam release for dose analysis. The licensee stated that the existing AORs already consider the steam released to remove the stored energy in the fuel, RCS, and steam generators; core decay heat; and changes in RCS and steam generator fluid masses and associated conditions. Therefore, the licensee indicated that its assessment determined the total energy to be removed is from the additional decay heat from Co-60 production. The licensee provided the calculated energy balance for the intact steam generators. Table 8.8-1, "Salem Steamline Break Releases," and table 8.8-2, "Salem Locked Rotor Releases," of attachment 3 to the LAR provide the AOR releases for the steam line break and locked rotor events and present the steam release from Co-60 production decay heat. The results show that the increased steam release from Co-60 production for both events, from 2 to 32 hours, would be []

[] over the current AOR results in the UFSAR. The licensee determined that rod ejection releases would not be impacted by the small increase in decay heat from Co-60 production.

The NRC staff reviewed the licensee's assessment and determined that the licensee's energy balance calculation presented in the LAR appropriately considered various inputs and provides reasonable assurance that the decay heat contribution from Co-60 production has minimal impacts on the steam release for dose analysis.

3.6.9 Evaluation of Radiological Consequences from Normal Operations, Co-60 Harvesting, and DBAs

The LAR provides the licensee's assessment of the impact of Co-60 production on radiation protection during normal operations and the radiological consequences of DBAs. The NRC staff reviewed the proposed changes and licensee assessments against the regulations in 10 CFR Part 20 and 10 CFR 50.67, the guidance in RG 1.183, and NRC's prior approval of the AST methodology for Salem, Unit Nos. 1 and 2 (Reference 5).

3.6.9.1 Radiation Protection During Normal Operations

The NRC staff evaluated the radiation protection implications of introducing COBA assemblies during normal reactor operations pertaining to the Co-60 production process described throughout the LAR. The NRC staff focused its review on potential changes in routine radiological source terms, RCS Co-60 concentrations, plant radiation monitoring and chemistry programs, and potential worker dose impacts during normal operation.

3.6.9.1.1 Reactor Coolant Co-60 Activity and Chemistry Controls

Section 4.10 of attachment 3 to the LAR describes the licensee's assessment of the potential for cobalt release from a COBA capsule and compares it to historical RCS Co-60 activity levels. The licensee's assessment concluded that:

- The breach of a COBA capsule is very unlikely.
- In the unlikely event that a breach occurred, the release would be gradual.
- RCS Co-60 concentration has been stable for over 10 years, with observed concentrations far below the values in table 11.1-8, "Reactor Coolant Equilibrium Fission and Corrosion Product Activities," of the UFSAR.
- The plant chemistry program already monitors Co-60 routinely.
- The licensee has determined an administrative equilibrium threshold value for Co-60 well below levels used in current licensing basis dose analyses that would trigger follow up investigations to ensure that current licensing basis values remain bounding.

The NRC staff determined that the licensee's assessment and ongoing chemistry and radiological monitoring provide reasonable assurance that any unexpected changes in RCS activity related to Co-60 would be identified and addressed. The NRC staff finds that because of the insignificant contribution of the Co-60 capsules to the RCS activity, there would not be a significant contribution to operational worker dose.

3.6.9.1.2 Offsite Dose Impacts

Attachments 2 and 3 to the LAR provide no indication of increased gaseous or liquid effluent releases from Co-60 production during normal operations. Specifically, COBA assemblies are encapsulated and introduce no new pathways for routine radionuclide release, and there is no impact on offsite dose calculations. Based on this information, the NRC staff concludes that offsite doses during normal operation would not change.

3.6.9.1.3 Harvesting Team Dose Evaluation

Section 10, "Harvesting," and section 11, "Measurement of Cobalt-60 Activity," of attachment 2 to the LAR describes the harvesting process and activity measurement system (AMS) used to extract and prepare for shipment irradiated COBA capsules. Section 11.2 of attachment 2 to the LAR indicates that the capsule activity confirmation process is controlled in accordance with the NRC-approved Westinghouse Quality Management System. Calculations performed by the licensee indicate routine area dose rates during refueling and harvesting will be [[

]]

Handling of loaded transport casks is performed under water with expected external dose rates of less than 10 mrem per hour during cask preparation. These activities will be procedurally controlled, with foreign material capture and [[

]]

Based on the information provided, the NRC staff determined that the estimated collective dose of [[]] for the harvesting campaign is well below the annual adult occupational TEDE limit in 10 CFR 20.1201 of 5 rem. The NRC staff determined that [[]] does not materially change access controls or require modifications to plant shielding or monitoring systems. The NRC staff determined that the licensee's administrative controls on [[]] and foreign material control would further bound exposure opportunities.

The NRC staff finds that the licensee's estimates and described controls provide reasonable assurance that occupational doses to the harvesting team will remain within 10 CFR 20.1201 limits. Therefore, the NRC staff concludes that the proposed harvesting operations associated with the COBA program are acceptable from an occupational dose perspective.

3.6.9.1.4 Conclusion on Radiation Protection Acceptability During Normal Operations

Based on its review of the information provided by the licensee, the NRC staff determined the following:

1. The introduction of COBA assemblies does not adversely affect radiation protection conditions during normal power operations.
2. Existing chemistry, radiological monitoring, and radiation protection programs are sufficient to manage any small changes in Co-60 inventory.
3. No new radiation protection hazards or regulatory nonconformances are introduced during normal operation.

Therefore, the NRC staff finds that the radiation protection aspects of the proposed LAR during normal operations are acceptable.

3.6.9.2 Radiological consequences of DBAs

Section 8.9, "Radiological Consequences (Doses) of Accidents," of attachment 3 to the LAR describes the licensee's assessment of the impacts of Co-60 production on the doses assessed to occur at the exclusion area boundary (EAB), low population zone (LPZ), and control room during the following DBAs: LOCA, MSLB, and a reactor coolant pump (RCP) locked rotor accident (LRA).

3.6.9.2.1 Licensee's Methodology

Section 8.9 of attachment 3 to the LAR describes the licensee's assessment of the impacts of Co-60 production on the doses assessed to occur at the EAB, LPZ, and control room following a LOCA, MSLB, RCP LRA. The licensee developed a mechanistic Co-60 release fraction for severe-accident conditions based on the capsule and slug design and assessed the LOCA dose impact using that release fraction against RG 1.183. The licensee reassessed these DBAs using an updated core Co-60 inventory and small decay heat additions attributable to activated COBA assemblies.

The licensee stated that a post-accident release of Co-60 from COBA fuel inserts is not expected based on the design details and analyses described in section 8.6.1 of attachment 3 to the LAR. The licensee concluded that the dose impact of the added Co-60 in the COBA assemblies on the LOCA dose analysis is expected to be small.

The NRC staff determined that the COBA Co-60 release fraction used in the licensee's reassessed LOCA dose analysis [

]] The licensee indicated that the resulting dose impacts are minimal, as shown in the reassessments described in the LAR.

The licensee also evaluated the small additional decay heat contribution from activated COBAs and its effect on steam releases for the MSLB and reactor coolant pump locked rotor events in section 8.9 of attachment 3 to the LAR. The licensee quantified the incremental steam release attributable to COBA decay heat and showed that the overall dose consequences remain essentially unchanged. For each DBA considered, the licensee compared the current doses to revised doses that include the Co-60 production impacts on source term and steam release.

3.6.9.2.2 Loss-of-Coolant Accident

The licensee reassessed post-LOCA doses at the EAB, LPZ, and control room with COBA impacts. Table 8.9-1, "Post-LOCA EAB, LPZ, and Control Room Dose Consequence Results," in attachment 3 to the LAR provides the licensee's calculated doses in comparison to its CLB values and regulatory requirements. The NRC staff determined that the licensee's results (also shown in table 1 below) show minimal increases in TEDE at the three receptor locations compared to current licensing basis values and remain below 10 CFR 50.67 limits.

Table 1: Comparison of Current LOCA Doses, Revised LOCA Doses with COBA, and Regulatory Limits

Location	CLB*	With COBA*	10 CFR 50.67*
EAB	6.86	6.91	25
LPZ	1.21	1.21	25
Control Room	4.33	4.35	5

* Values are in rem.

3.6.9.2.3 Main Steam Line Break

The licensee reassessed MSLB doses for both pre-accident iodine spike and concurrent iodine spike cases for both Salem units. The NRC staff determined that the results shown in table 8.9-2, "Unit-1 Main Steam Line Break EAB, LPZ and CR Dose Consequence Results," and table 8.9-3, "Unit-2 Main Steam Line Break EAB, LPZ and CR Dose Consequence Results," of attachment 3 to the LAR indicate no meaningful change from COBA-related increases in decay heat and associated steam releases, and that MSLB accident doses remain below the acceptance criteria for MSLB cases specified in table 6, "Accident Dose Criteria," of RG 1.183.

3.6.9.2.4 Reactor Coolant Pump Locked Rotor Accident

The licensee reassessed locked rotor doses for both units. The NRC staff determined that the results shown in table 8.9-4, "Locked Rotor Accident Dose Consequence Results," of

attachment 3 to the LAR show minimal increases that continue to maintain large margins compared to the acceptance criteria in table 6 of RG 1.183.

3.6.9.2.5 Conclusion

Based on its review of the licensee's analyses against the applicable regulatory guidance in RG 1.183, the NRC staff finds reasonable assurance that the radiological consequences of the analyzed DBAs with COBA assemblies meet the acceptance criteria specified in 10 CFR 50.67 and table 6 of RG 1.183 and, therefore, are acceptable from a dose consequence perspective. The NRC staff determined that the licensee has demonstrated that the proposed Co-60 production will not result in a significant increase in accident consequences.

3.6.10 Evaluation of Heat Load on Cooling Water Flow Requirements

The paragraph titled, "Decay Heat from COBAs," in section 8.2 of attachment 2 to the LAR states that the COBA rod component heating created by the gamma radiation produced by the decay of Co-60 has been determined for use in the safety analyses. The licensee assumes that all heat generated by the Co-60 decay is deposited into the core as an additional component of decay heat. The licensee considers a conservative total value has been calculated that bounds the decay heat added by the inclusion of COBA in the core design. The licensee assumes that the added decay heat component from Co-60 decay remains constant for all safety analyses due to the relatively long Co-60 half-life. The licensee asserts that the total decay heat for each core will be ensured to remain below the calculated conservative total value by calculating the total Co-60 Curie production in a given cycle and comparing it to the calculated total decay heat from COBA that is included in safety analysis evaluations.

The NRC staff audited the licensee's supporting analyses demonstrating how 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 SFP. In the "Audit Item 24" section of its supplement dated June 24, 2026, the licensee stated that the ability to remove increased decay heat from the RCS was accounted for in the RHR cooldown analysis assessment for Co-60. This analysis assessed the impact of the additional decay heat on the ability to cool the plant in various scenarios. The licensee evaluated the RCS coolant flow through the core with respect to core bypass flow. The licensee determined that the inclusion of COBAs in the core will have no adverse effect on the core bypass flow fraction and core pressure drop. The maximum decay heat added to the core during operation is very small compared with the licensed core power of 3,457 MWt. The licensee determined that an increase in heat in the core from the COBA decay heat would result in an insignificant impact on the vessel outlet temperature.

In the "Audit item 26" section of its supplement dated June 24, 2026, the licensee provided its evaluation of SFP cooling from the introduction of COBA fuel inserts, including the impact of the increased decay heat from the COBAs. Section 9.1.3.2, "Spent Fuel Pool System – System Description," of the UFSAR describes the SFP integrated decay heat management program (IDHMP), which requires a pre-outage assessment of the SFP heat loads and heat up rates to assure available SFP cooling capability prior to offloading fuel. Prior to each core offload, the decay heat management program methodology is used to (1) calculate that the water temperature will not exceed 149 °F following full core offload, using only one heat exchanger for each SFP and to provide to the plant staff the required component cooling water

temperature to achieve such results; and (2) calculate that the water temperature will not exceed 180 °F following full core offload with one heat exchanger available for both SFPs and provide to the plant staff the required component cooling water temperature to achieve this result. The licensee stated that prior to core offload, it updates the plant thermal-hydraulic calculation for that specific core offload using the fuel assemblies in the SFP and the reactor core. The licensee stated that the COBA decay heat will be included as a fixed heat load for the time span of this calculation. As a result, the impact on SFP cooling from the additional decay heat from COBA fuel inserts is addressed in the plant-specific calculation.

Based on its review, the NRC staff finds that the licensee has demonstrated that the safety systems will not be adversely impacted by the additional heat load from the COBAs.

3.7 Evaluation of Effects on Spent Fuel Pool

3.7.1 Evaluation of Spent Fuel Pool Criticality

Section 9.1, “Spent Fuel Pool Criticality,” of attachment 3 to the LAR describes the licensee’s assessment of the impact of COBA inserts on the CSA for the SFP and NFSV. The major aspects of the CSA that were changed to address the storage of COBA fuel assemblies included: the explicit modeling of COBA fuel assemblies, including the resultant effects on isotopic composition and associated axial profiles, and the inclusion of COBA fuel assemblies in criticality fuel designs within the CSA. Because of the scope of the updated CSA, it was submitted as a separate LAR and approved by the NRC on June 26, 2026 (Reference 7).

As stated in section 9.1 of attachment 3 to the LAR, the licensee concluded there was no impact on NFSV criticality from the COBA. The licensee stated that no credit was taken by the poison contribution of the COBA fuel inserts, and that the unirradiated COBA inserts have no impact on the fuel assembly isotopic compositions and associated axial profiles. The licensee stated that this approach has precedent in that the same treatment was followed for new WABAs.

Based on its review of the information in the LAR and NRC’s prior approvals of the WABAs (References 8 and 9) and the revised CSA, the NRC staff finds the inclusion of COBA inserts does not negatively impact the CSA for the storage of fuel assemblies in the SFP and NFSV.

3.7.2 Evaluation of Spent Fuel Pool Time to Boil

Section 9.2, “Spent Fuel Pool Time to Boil,” of attachment 2 of the LAR describes the licensee’s assessment of the impact of COBA inserts on SFP heat up behavior and time to boil (TTB). The licensee indicated that its analysis modeled two scenarios and assumed that the decay heat produced by the COBA inserts would be transferred directly to the SFP water. The licensee’s assessment resulted in small increases in total decay heat from the COBA inserts and, therefore, a slight reduction in the TTB for both scenarios.

The NRC staff reviewed the licensee’s assessment of the impact of COBA inserts on SFP heat up behavior and TTB described in section 9.2 of attachment 2 of the LAR. The NRC staff finds that the reductions in TTB are small relative to the conservative uncertainty included in the decay heat model and do not challenge SFP temperature limits, heat up assumptions, or cooling capability. The licensee’s methodology credits cycle-specific thermal evaluations prior to moving spent fuel into the SFP that appropriately accounts for the COBA decay heat contribution. The NRC staff determined that the COBA decay heat impacts on the SFP TTB are

in conformance with the SFP heat load management program described in section 9.1.3.2 of the UFSAR. Based on its review of the information provided in the LAR, the NRC staff concludes that the introduction of COBA inserts does not adversely affect the SFP TTB and, therefore, is acceptable.

3.7.3 Evaluation of Gamma Heating of the SFP Concrete Structure

Section 9.3, "Gamma Heating of the SFP Concrete Structure," of attachments 2 and 3 to the LAR describes the licensee's assessment of the potential for degradation of the SFP concrete walls and floor from the anticipated increase in the incident gamma energy flux from storing COBA fuel assemblies and conducting COBA rodlet-harvesting activities. The NRC staff reviewed the licensee's acceptance criteria, primary design inputs, assessment results, and conclusions.

3.7.3.1 Evaluation of Acceptance Criteria and Primary Design Inputs

Section 9.3.1, "Acceptance Criteria," of attachments 2 and 3 to the LAR describes the acceptance criteria involving energy flux and dose that the licensee used in its assessment, which are derived from section 4.3.1.1 of NUREG/CR-6927 and section A.4.1 of ACI 349-01. Section 9.3.2, "Primary Design Inputs," of attachments 2 and 3 to the LAR describes the key assumptions and potential Co-60 harvesting and temporary storage configuration scenarios used as inputs in its analysis of gamma heating of the SFP concrete structure.

The NRC staff reviewed the licensee's acceptance criteria and confirmed that the energy flux and dose threshold values used for the licensee's acceptance criteria are consistent with the guidance in NUREG/CR-6927, which establishes these values as limits below which gamma-radiation effects on SFP concrete structures are not expected to occur. Furthermore, the NRC staff determined that licensee's use of the ACI 349-01 temperature limits provides an appropriate means to evaluate when concrete heating may be non-negligible once the incident energy-flux threshold is exceeded. Based on this review, the NRC staff finds that the licensee's acceptance criteria are acceptable for determining whether the incremental gamma flux and integrated dose from COBA fuel assemblies and COBA rodlet-harvesting activities could adversely affect the SFP concrete structures.

The NRC staff reviewed the information in section 9.3.2 of attachments 2 and 3 to the LAR and determined that the licensee's key design inputs associated with the COBAs and spent fuel pool and the intended operational procedures associated with the Co-60 harvesting process are acceptable because the design inputs establish clear, bounded, and technically supported parameters for COBA configuration and harvesting operations.

3.7.3.2 Evaluation of Assessment Results and Conclusions

Section 9.3.3, "Gamma Heating of the SFP Concrete Structure," and section 9.3.4, "Conclusion," of attachment 3 to the LAR describe the results and conclusions of the licensee's assessment of the effects of COBA assembly storage and harvesting activities on the SFP

concrete walls and floor. In section 9.3.3 of attachment 3 to the LAR, the licensee concluded the following:

- By meeting placement requirements, the COBA fuel assemblies could remain in the SFP for the remaining life of the plants without exceeding the ACI 349-01 concrete temperature limits or irradiation limits of 10^{10} radiation absorbed dose (rads).
- The contribution of the COBA rodlets in the COBA fuel assemblies to the total flux at the SFP floor is essentially insignificant when compared to the NUREG/CR-6927 criterion of less than 10^{10} MeV/cm²-sec.
- The potential for concrete degradation from irradiation during the harvesting process over the remaining life of the plant will remain within the irradiation limits of 10^{10} rads.

In section 9.3.4 of attachment 3 to the LAR, the licensee stated that prior to movement of irradiated COBA fuel assemblies from the core to the SFP, administrative controls will determine if the licensee's assessment remains bounding or whether an outage-specific analysis needs to be developed to maintain the incremental incident gamma flux on concrete surfaces from storage of a cluster of COBA fuel assemblies to within the specified criterion in NUREG/CR-6927 of less than 10^{10} MeV/cm²-sec and less than 10^{10} rads, such that nuclear heating and irradiation of concrete structures, respectively, need not be considered. The licensee also stated that administrative controls will ensure that COBA rodlet harvesting activities are not permitted during refueling and core-offload scenarios and when the temperature of the SFP water exceeds 120 °F.

The NRC staff's evaluation of this information is provided below.

3.7.3.2.1 Evaluation of Gamma Influx and Gamma Dose Assessment Methodologies

As described in section 9.3.3 of attachments 2 and 3 to the LAR, the licensee assessed the impacts from gamma heating on the SFP concrete structure by assessing both the thermal impacts (by using energy flux or temperature criteria) and the dose impacts. The licensee's assessment established placement requirements and critical distances at which acceptable results could be assured for both acceptance criteria by considering (1) conditions where COBAs are inserted in fuel assemblies stored in the spent fuel pool and (2) conditions during the harvesting process and temporary storage of source material prior to shipment. The licensee indicated that COBAs contribute very little to the energy flux on the SFP floor. The licensee indicated that compliance with critical distances during COBA harvesting and source material storage would provide assurance that adverse thermal and dose impacts associated with the COBAs on spent fuel pool concrete structures would be avoided.

In section 9.3.4 of attachment 2 to the LAR, the licensee indicated that no adverse impacts to the SFP structure would occur from the thermal or dose impacts resulting from introducing COBAs, provided that the licensee adheres to its proposed guidance addressing (1) storage of COBA fuel assemblies and (2) harvesting and storage of COBA rodlets and source material. In section 9.3.4 of Attachment 3 to the LAR submittal, the licensee indicated that, prior to movement of irradiated COBA fuel assemblies to the spent fuel pool, administrative controls would determine whether the evaluation described in the LAR submittal remains bounding or whether an additional outage-specific analysis must be developed to demonstrate that the

incremental incident gamma flux on concrete surfaces remains within the criteria for addressing thermal and irradiation effects on concrete. The licensee also indicated that administrative controls will ensure that COBA harvesting activities are not conducted during refueling or core-offload evolutions or when the temperature of the SFP coolant exceeds 120 °F.

The NRC staff found that the licensee's application of the energy flux (i.e., 10^{10} MeV/cm²-sec) and dose (i.e., 10^{10} rads) acceptance criteria specified in NUREG/CR-6927 in determining critical distances for six of the eight cases analyzed in its LAR provides reasonable assurance of preventing adverse irradiation impacts on the SFP concrete walls and floor. Based upon the licensee's analysis demonstrating consistency with the NUREG/CR-6927 criteria, the NRC staff finds the licensee's approach for maintaining critical distances in excess of the minimum calculated requirements to be acceptable for these six cases.

As discussed in section 9.3.3 of attachment 3 to the LAR, for two cases (i.e., cases 1 and 8) involving the harvesting process and temporary storage prior to shipment of the cobalt source material, the critical distances calculated by the licensee did not satisfy the acceptance criteria in NUREG/CR-6927. However, for these two cases, the licensee applied the temperature requirements in section A.4.1 of ACI 349-01 as acceptance criteria instead. The following subsections of this safety evaluation describe the NRC staff's evaluation of these two cases.

3.7.3.2.2 Evaluation of the SFP Concrete Structures Temperatures with Potential Increase from Gamma Heating Effect

The NRC staff evaluated whether elevated temperatures in the SFP reinforced concrete structure, including any temperature increases attributable to gamma heating from irradiated COBA fuel assemblies or COBA rodlet-harvesting activities, could result in adverse structural effects. Section A.4.1 of ACI 349-01 establishes that concrete temperatures during normal operation or other long-term period shall not exceed a concrete bulk average temperature of 150 °F, except for local areas that are allowed to have increased temperatures not to exceed 200 °F.

Section 9.1.3.2 of the UFSAR indicates that the SFP water temperature is maintained below 149 °F under normal conditions and below 180°F under abnormal conditions. In the "Audit Items 26(a) and 26(b)" section of the supplement dated June 24, 2026, the licensee indicated that the maximum gamma flux incident on the SFP walls and floor from COBA fuel assemblies remains below the 10^{10} MeV/cm²-sec threshold in NUREG/CR-6927.

The NRC staff reviewed the licensee's gamma flux evaluations for COBA rodlet-harvesting activities in the cask pit area. Table 9.3-1, "Scenarios 1, 2 & 3 – Incident Gamma Flux vs Acceptance Criteria of $< 10^{10}$ MeV/cm²-sec," of attachment 3 to the LAR shows that all cases remain within NUREG/CR-6927 criteria except cases 1 and 8. As discussed in its supplement dated June 24, 2026, the licensee performed thermal analyses for these two cases using the GOTHIC computer code to develop the steady state temperature profiles in the cask pit wall or floor. For case 1, the licensee's analysis indicated a small local region of the cask pit wall reaching approximately [], while the concrete bulk average temperature remains below []. For case 8, the licensee's analysis indicates that the maximum local peak temperature is approximately []. The NRC staff finds that the concrete bulk average and localized temperatures for both cases meet the threshold limits specified in section A.4.1 of ACI 349-01. Based on its review, the NRC staff finds that there is reasonable assurance that

elevated temperatures, including those associated with gamma heating, have no adverse structural effects in the SFP and the cask pit walls and floor.

3.7.3.2.3 Evaluation of Irradiation Effects on SFP Concrete: Reduction of Strength and Mechanical Properties Due to Integrated Gamma Dose

The NRC staff reviewed the information provided in the “Audit Item 26(a) and 26(b)” section of the supplement dated June 24, 2026, and determined that the integrated gamma dose to SFP walls and floor from COBA fuel assemblies and to cask pit walls and floor from the COBA rodlet-harvesting activities remains below the 10^{10} rads threshold in NUREG/CR-6927. Based on its review, the NRC staff finds that there is reasonable assurance that the integrated gamma dose associated with the COBA fuel assemblies and the COBA rodlet-harvesting activities does not adversely affect the strength or mechanical properties of the SFP or the cask pit walls and floor.

3.7.3.2.4 Evaluation of Structural Integrity of SFP Concrete Structures

The NRC staff reviewed the information provided in the “Audit Item 33” section of the supplement dated June 24, 2026, in which the licensee assessed the structural, seismic, and thermal effects associated with storing fuel assemblies containing COBA inserts in the SFP. Item 7 in section 3.8.4.4 of the UFSAR indicates that the SFP concrete structure, stainless steel liner, and liner anchors demonstrate compliance with ACI 359-95 (ASME BPV Code, Section III, Division 2), “Code for Concrete Reactor Vessels and Containments,” for both boiling pool conditions and a maximum normal pool temperature of 150°F. The licensee concluded in its supplement that the existing analyses in item 7 of section 3.8.4.4 of the UFSAR remain bounding and are not impacted by COBA implementation for the following reasons:

- Structural loading from COBA inserts is bounded by existing non-fuel components (i.e., RCCAs and WABAs that are already qualified for storage in the SFP racks).
- The seismic behavior of a fuel assembly containing a COBA fuel insert is bounded by the seismic behavior of fuel assembly with RCCAs or WABAs.
- Thermal loading is unchanged because the relevant requirements pertain to water temperature limits for specified SFP configurations, and the decay heat contribution from the COBA fuel inserts is included in the water temperature analysis.

The NRC staff’s evaluation of gamma heating effect on the SFP concrete structures is documented in section 3.7.3.2.2 of this safety evaluation.

In the “Audit Item 33” section of the supplement dated June 24, 2026, the licensee discussed the temporary harvesting workstation to be used during COBA harvesting campaigns. The licensee indicated that the workstation would be installed only for the duration of a harvesting campaign and removed once harvesting activities are complete, and that COBA harvesting does not involve any permanent plant modifications. The licensee also indicated that the structural and seismic loadings associated with the workstation are bounded by the existing design-basis evaluations for dry cask storage operations conducted in the same cask pit location.

The NRC staff reviewed the licensee's evaluation of potential gamma heating effects associated with COBA harvesting activities. The licensee assessed the potential increase in gamma-heating due to the higher gamma flux on the cask pit walls and floor during COBA harvesting campaigns compared to dry cask storage campaigns. The NRC staff's evaluation of SFP concrete structure temperatures, including consideration of potential increases from gamma-heating, is documented in section 3.7.3.2.2 of this safety evaluation. Based on its review, the NRC staff finds that the harvesting workstation does not adversely affect the cask pit structural analyses and that its use is neither affected by nor related to COBA implementation.

Based on its review, the NRC staff finds that there is reasonable assurance that the structural integrity of the SFP concrete structure, stainless steel liner, and liner anchors will continue to be maintained under conditions associated with the storage of COBA fuel assemblies and the temporary installation of the harvesting workstation and, therefore, the SFP structure will continue to conform to of AEC proposed GDC 2, as described in the UFSAR, for withstanding the effects of natural phenomena without loss of safety function.

3.7.4 Spent Fuel Pool Cooling

Section 9.4, "SFP Cooling," of attachments 2 and 3 of the LAR describes the licensee's assessment of the added Co-60 decay heat on the SFP cooling capability. The licensee indicated that the added Co-60 decay heat from the COBA assemblies would only affect operations during an outage and that the added heat is a small fraction of the total decay heat from the fuel and, therefore, is expected to have a small impact on SFP cooling. As discussed in section 3.6.10 of this safety evaluation and in the "Audit Item 24" and "Audit Item 26" sections of the supplement dated June 24, 2026, the licensee described how it addressed 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 RCS and SFP to support the proposed Co-60 production.

Section 9.4 of attachment 3 to the LAR and the "Audit Item 26" section of the supplement dated June 24, 2026, indicate that the licensee's SFP IDHMP, as described in section 9.1.3.2 of the UFSAR, provides for the evaluation of SFP decay heat loads prior to each core offload. The IDHMP requires a pre-outage assessment of the SFP heat loads and heat up rates to assure available SFP cooling capability prior to offloading fuel. The licensee indicated that prior to each core offload, the decay heat management program methodology is used to: (1) calculate that the water temperature will not exceed 149 °F following full core offload, using only one heat exchanger for each SFP and to provide to the plant staff the required component cooling water temperature to achieve such results; and (2) calculate that the water temperature will not exceed 180 °F following full core offload with one heat exchanger available for both SFPs and provide to the plant staff the required component cooling water temperature to achieve this result. The licensee stated that prior to core offload, the plant engineering staff update the plant thermal-hydraulic calculation for that specific core offload using the fuel assemblies in the SFP and the reactor core. The licensee indicated that the COBA decay heat will be included as a fixed heat load for the time span of this calculation and, as a result, the impact on SFP cooling from the additional decay heat from COBA fuel inserts is addressed in the plant-specific calculation.

The NRC staff evaluated the possible consequences of the additional heat to the SFP from the COBAs and determined that the SFP cooling system and the IDHMP have sufficient capacity

and capability to handle the increased decay heat introduced by the COBA assemblies. Based on its review, the NRC staff finds that the licensee has demonstrated that the SFP cooling will not be adversely impacted by the additional heat load from the COBAs.

3.8 Evaluation of Harvesting

The NRC staff reviewed the information provided in sections 10.1 through 10.5 of attachment 2 to the LAR covering the harvesting process, workstation tooling, foreign material control, and waste disposal methods. In section 10.1, "Harvesting Process Overview," of attachment 2 of the LAR, the licensee stated that the harvesting system is used to [[

]]

The NRC staff considered the potential impacts to safety should the harvester not perform as designed, including foreign material from the cutting process, loss of physical control of a capsule, and failure to maintain the digital serialization that preserves the identity of the capsules. The NRC staff determined that none of these potential consequences would have a significant impact on safety. The most significant concern, foreign material generated during cutting, is mitigated through the use of a [[]].

The NRC staff previously reviewed (Reference 12) the digital serialization methodology described in the Westinghouse letter, "Submittal of the Westinghouse Cobalt-60 Digital Serialization Approach White Paper" (Reference 10), as supplemented (Reference 11), including the use of the harvesting system to establish and maintain the association between harvested capsules and their assigned digital serial numbers. As documented in the NRC staff's review, the NRC staff concluded that the proposed methodology provides for unique identification of the harvested capsules. The harvesting process described in the licensee's LAR is consistent with the information previously reviewed with respect to digital serialization. The NRC staff therefore relies on its previous evaluation for the digital serialization aspects of the harvesting process and concludes that the harvesting process is acceptable because failures of the serialization scheme or of the harvester itself do not pose a threat to public health and safety.

Section 3.7.3 of this safety evaluation documents the NRC staff's evaluation of section 10.6, "Effects Within Spent Fuel Pool," of attachment 2 to the LAR regarding gamma heating effects on the SFP concrete structures. Section 3.6.9 of this safety evaluation documents the NRC staff's evaluation of section 10.7, "Exposure to Personnel," of attachment 2 to the LAR regarding harvesting operations impacts on personnel dose.

3.9 Evaluation of Measurement of Cobalt-60 Activity

Section 11, "Measurement of Cobalt-60 Activity," of attachment 2 to the LAR describes how the licensee would measure Co-60 activity in preparation for shipping. The licensee described [[]] source cage that can hold 52 capsules. Figure 9.3-4, "Source Cage" of attachment 3 to the LAR shows the configuration of each source cage with 52 capsule positions. Section 9.3.2, "Primary Design Inputs," of attachment 2 to the LAR explains each Nordion F-231 transport cask is intended to transport two source cages. Section 9.3.2 also states that each capsule will contain [[]] slugs, and each slug contains up to a maximum of [[]] of Co-60, for a total of [[]] of Co-

60 per capsule. Therefore, as described by the licensee, if a single source cage were fully loaded (i.e., if all 52 positions in a source cage were filled with a fully-irradiated capsule), a single source cage could contain up to [] of Co-60, and two source cages loaded into the F-231 transport cask, absent an administrative limit, could contain up to [] of Co-60.

Section 9.3.2 of attachment 2 to the LAR indicates that to ensure the Nordion F-231 transport package limit of 400,000 Ci is not exceeded, the curie content for each of the two source cages transported in a Nordion F-231 cask is administratively restricted to 180,000 Ci, and the total curie content for a transport cask, which is able to transport two source cages, is likewise limited. Additionally, section 10.2.4 of attachment 2 to the LAR indicates that []

[]

The NRC staff verified that the Nordion F-231 transport package (USA/0578/B(U)-96, Revision 7) is currently certified by Canada and approved for use within the United States for import and export shipments made in accordance with applicable international and domestic transport regulations (Reference 33). This cask has vent and drain lines to facilitate wet loading that are plugged by safety cables and capped. The NRC staff verified that the authorized contents of the F-231 transport package include, among other things, not more than [] that meet the requirements of ISO 2919:2012. The requirements of ISO 2919:2012 are summarized in section 3.2.7 of this safety evaluation. As described in the package certificate, the average surface heat flux of the package with [] of Co-60 is [] [] The certificate also specifies that for heat fluxes exceeding [] [] supplementary arrangements must be made with the carrier to ensure adequate heat dissipation.

The NRC staff evaluated the licensee's proposed description of how it will ensure that the total activity does not exceed the licensed limit of the shipping cask prior to shipment and determined that the proposed measures provides reasonable assurance that . the licensee will not exceed its proposed []

[]

3.10 Evaluation of Byproduct Material Tracking and Reporting

In section 12, "Byproduct Material Tracking and Reporting," of attachment 2 to its LAR, the licensee described its approach to byproduct material tracking and reporting for Co-60 produced in the reactor. The licensee stated, "... these sources do not fall within the scope of the nationally tracked source reporting requirements in 10 CFR 20.2207. Consequently, they would also not be subject to the serialization requirements in 10 CFR 32.201."

The NRC staff previously evaluated this issue in its review of the licensee's white paper on digital serialization, as supplemented (References 10 and 11) and documented its position in a staff response dated July 23, 2025 (Reference 12). In that evaluation, the staff considered the licensee's proposed production process and handling configuration and concluded, "... the NRC staff considers these sources to fall outside the scope of the nationally tracked source reporting

requirements in 10 CFR 20.2207. Consequently, they would also not be subject to the serialization requirements in 10 CFR 32.201.”

The NRC staff reviewed the licensee’s description of the harvesting process in section 10, “Harvesting,” of attachment 2 to the LAR and determined that the Co-60 material produced during harvesting is consistent with the production process previously evaluated in the white paper response. Specifically, the material remains an intermediate product intended solely for transfer to a source manufacturing licensee for further fabrication and is not in its final sealed source form for end use. Therefore, the basis for the NRC staff’s previous conclusion remains unchanged. Accordingly, the NRC staff concludes that the Co-60 material, as described in the LAR, falls outside the scope of the nationally tracked source reporting requirements in 10 CFR 20.2207 and is not subject to the serialization requirements in 10 CFR 32.201.

3.11 Evaluation of First Cycle Post-Irradiation Examination

In section 13, “First Cycle Post-Irradiation Examination,” of attachment 3 to the LAR, the licensee indicated that performance of post-irradiation examination of the COBAs would be considered optional. In its supplement dated June 24, 2026, the licensee submitted information concerning the pedigree of the components used to fabricate the COBAs and the expected efficacy of visual inspections. In the “Audit Item 18” section of its supplement dated June 24, 2026, the licensee indicated that concerning the COBAs, while the integrated design may be considered new, extensive experience exists relative to the components from which they are constructed that provides reasonable assurance of COBA integrity under postulated conditions:

- The structure of the COBA, which involves rodlets inserted into fuel assembly thimble tubes, draws significantly upon previous Westinghouse WABA designs. Similar materials and methods for handling, assembly, and disassembly are also employed.
- The design, fabrication, and inspection of the encapsulated cobalt targets draws upon decades of experience Nordion and its supply-chain partners have accumulated, mainly at Canadian deuterium uranium (CANDU) reactors.

Although the COBA design itself is novel, the NRC staff determined that the combination of mature WABA mechanical experience, decades of cobalt capsule irradiation experience in CANDU reactors, and COBA specific design analyses provide confidence that the COBAs would behave as expected. Furthermore, (1) routine monitoring of cobalt-60 activity in the reactor coolant would provide indication of unexpected degradation and (2) refueling shuffles provide opportunities to directly observe physical condition without disassembly. The licensee indicated in attachment 3 to the LAR and in its supplement dated June 24, 2026, that it would consider performing post-irradiation examination if there are indications of: (1) unexpected cobalt-60 increases in coolant occur, or (2) abnormal conditions are observed during refueling handling. In light of the multi-layered containment of the radioactive source material, stringent fabrication and inspection, engineered mechanical robustness, and in-operation monitoring, the NRC staff determined that the proposed post-irradiation examination approach is acceptable and that it is not necessary to specify in advance criteria for determining when PIE should be performed.

3.12 Evaluation of Proposed Change to TS 5.3, "Reactor Core"

The regulation in 10 CFR 50.36(c)(4) requires that the design features to be included in the TS are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in the safety limits, limiting safety systems settings, limiting control settings, LCOs, and surveillance requirements sections of the TS.

In its supplement dated June 24, 2026, the licensee proposed to add TS 5.3.3, "Cobalt Burnable Absorber (COBA) Assemblies," to the "Reactor Core" section of the TSs to specify design features information related to COBA assemblies. As described in section 1.2 of this safety evaluation, the licensee proposed adding the new TS to authorize Salem to place a maximum of 288 cobalt burnable absorber rodlets in the reactor core in an operating cycle.

The NRC staff reviewed the proposed changes. The LAR, as supplemented, contained a substantial technical justification for the acceptability of installing a maximum number of COBAs in the reactor core. The NRC staff's technical evaluation in sections 3.1 through 3.11 of this safety evaluation determined that this change is acceptable and that all regulatory requirements are met. Based on this technical evaluation, the NRC concluded that the proposed TS 5.3.3 for each unit authorizing to place a maximum of 288 COBA rodlets in an operating cycle is necessary to ensure TS 5.3 continues to contain a description of the reactor core materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in safety limits, LCOs or surveillance requirements, as required by 10 CFR 50.36(c)(4). Therefore, the NRC staff finds the proposed change to add TS 5.3.3 to the TSs is acceptable because the provisions in 10 CFR 50.36(c)(4) will continue to be met.

3.13 Technical Evaluation Summary

The NRC staff determined that the COBA assembly design considerations and fabrication are acceptable and meet the regulatory evaluation criteria specified in section 3.1. The NRC staff determined that the COBA design criteria; the cobalt slug and capsule configuration, mechanical integrity, creep analysis, production leak test, and wear evaluation; COBA baseplate connections; and the licensee's assessment of COBA insertion on operation risk and capsule breach on reactor coolant chemistry are acceptable and meet the regulatory evaluation criteria specified in section 3.2 of this safety evaluation. The NRC staff determined that the thermal and hydraulic impacts of the proposed changes are acceptable. The NRC staff determined that the materials specifications, properties, and applications for the capsules, COBA rodlet, baseplate connection, and associated parts, and the nickel coating and cobalt diffusion properties are acceptable and meet the regulatory evaluation criteria specified in section 3.4. The NRC staff determined that the licensee's methodology and analysis of how the COBA inserts function in the reactor core and interact with other core components are acceptable. The NRC staff determined that it has reasonable assurance that the impacts of Co-60 production on plant safety analyses and radiological consequences will remain within regulatory requirements and the design bases in chapter 15 of the UFSAR. The NRC staff determined that it has reasonable assurance that the SFP structural integrity and cooling capability of the SFP will be maintained when storing COBAs. The NRC determined that the proposed process for harvesting Co-60 would not have a significant impact on safety and that the licensee has an adequate process for tracking and reporting the Co-60 produced in the reactor. The NRC staff determined that the

licensee would have adequate measures in place to measure Co-60 activity. The NRC staff determined that the optional performance of post-irradiation examination of the COBAs was acceptable. The NRC staff determined that the proposed changes to the TS are acceptable because the provisions in 10 CFR 50.36(c)(4) will continue to be met.

Based on the technical evaluation presented in sections 3.1 through 3.12 of this safety evaluation, the NRC staff determined that the license conditions proposed in the supplement dated June 24, 2026, are acceptable for operation at the current licensed power level of 3,459 MWt.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of New Jersey official on August 28, 2026 (Reference 34), of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This action relates to authorizations under 10 CFR Part 50 with respect to installation or use of a facility component. The NRC staff has determined that any ground disturbance is limited to previously disturbed areas. Additionally, the NRC staff has determined that the action involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, no significant increase in individual or cumulative public or occupational radiation exposure, and no significant increase in the potential for or consequences from radiological accidents. Finally, the NRC staff has determined that a categorical exclusion applies and that special circumstances under 10 CFR 51.22, "Categorical exclusions," are not present that would preclude reliance on the categorical exclusion. Accordingly, this action meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(d)(8). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment need be prepared in connection with the action.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 ABBREVIATIONS

σ	standard deviation
°C	degrees Celsius
°F	degrees Fahrenheit
ΔP	delta pressure
ACI	American Concrete Institute
Act	Atomic Energy Act of 1954, as amended
AEC	Atomic Energy Commission

II	II	II	II
ANC		Advanced Nodal Code	
ANS		American Nuclear Society	
ANSI		American National Standards Institute	
AOR		analysis of record	
ASME		American Society of Mechanical Engineers	
AST		alternative source term	
BDB		beyond-design-basis	
BPV		Boiler and Pressure Vessel	
CANDU		Canadian deuterium uranium	
Ci		curies	
CFD		computational fluid dynamics	
CFR		<i>Code of Federal Regulations</i>	
CLB		current licensing basis	
CNSC		Canadian Nuclear Safety Commission	
COBA		cobalt burnable absorber	
COCO		Containment Pressure Analysis Code	
COMS		cold overpressure mitigation system	
CR		control room	
CSA		criticality safety analysis	
Co-59		cobalt-59	
Co-60		cobalt-60	
DBA		design basis accident	
DNB		departure from nucleate boiling	
DNBR		departure from nucleate boiling ratio	
DOT		U.S. Department of Transportation	
EAB		exclusion area boundary	
ELAP		extended loss of alternating current power	
EPRI		Electric Power Research Institute	
EQ		equipment qualification	
$F_{\Delta H}$		nuclear enthalpy rise hot channel factor	
FLEX		flexible coping strategies	
FMEA		failure modes and effects analysis	
GDC		General design criteria (criterion)	
IDHMP		integrated decay heat management program	
II	II	II	II
ILRT		integrated leak rate test	
ISO		International Organization for Standardization	
II	II	II	II
kGy/hr		Kilogray per hour	
LAR		license amendment request	
LBLOCA		large-break loss-of-coolant accident	
lbm		pounds-mass	
LCO		Limiting condition(s) for operation	
LOCA		loss-of-coolant accident	
LPZ		low population zone	
LRA		locked rotor accident	

LTOPS	low temperature overpressure protection system
M&E	mass and energy
MAAP	Modular Accident Analysis Program
MeV/cm ² -sec	megaelectron volts per square centimeter-seconds
mrem	milli-roentgen equivalent man (mrem)
MSLB	main steam line break
MWt	megawatts thermal
NCC	natural circulation cooldown
NFSV	new fuel storage vault
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NSAL	nuclear safety advisory letter
NSHC	no significant hazards consideration
NSSS	nuclear steam supply system
ORAM	Outage Risk Assessment and Management
PCT	peak cladding temperature
psia	pounds per square inch absolute
rad	radiation absorbed dose
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCS	reactor coolant system
RG	regulatory guide
rem	roentgen equivalent man
RHR	residual heat removal
RSAC	reload safety analysis checklist
RSE	reload safety evaluation
SBLOCA	small-break loss-of-coolant accident
[[]]	[[]]
SFP	spent fuel pool
SGTR	steam generator tube rupture
[[]]	[[]]
SRP	standard review plan
[[]]	[[]]
TEDE	total effective dose equivalent
THRIVE	Thermal Hydraulic Reactor Internal Vessel Evaluation
TS	Technical specification(s)
TTB	time to boil
UFSAR	Updated Final Safety Analysis Report
W/m ²	watts per square meter
WABA	wet annular burnable absorber
[[]]	[[]]
WTDP	Westinghouse Thermal Design Procedure
[[]]	[[]]

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SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 – ISSUANCE
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