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

October 8, 2021

Mr. Eric Carr
President and Chief Nuclear Officer
PSEG Nuclear LLC – N09
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 339 RE: REVISE AND RELOCATE REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS AND PRESSURIZER OVERPRESSURE PROTECTION SYSTEM LIMITS TO PRESSURE AND TEMPERATURE LIMITS REPORT (EPID L-2020-LLA-0263)

Dear Mr. Carr:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 339 to Renewed Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1, in response to your application dated December 6, 2020, as supplemented by letters dated March 17, 2021, April 1, 2021, and August 4, 2021.

The amendment revised the reactor coolant system pressure-temperature limits and the pressurizer overpressure protection system limits and relocated them to a pressure and temperature limits report.

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

Sincerely,

/RA/

James S. Kim, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-272

Enclosures:
1. Amendment No. 339 to DPR-70
2. Safety Evaluation

cc: Listserv

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 339 RE: REVISE AND RELOCATE REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS AND PRESSURIZER OVERPRESSURE PROTECTION SYSTEM LIMITS TO PRESSURE AND TEMPERATURE LIMITS REPORT (EPID L-2020-LLA-0263) DATED OCTOBER 8, 2021

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

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 339
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 dated December 6, 2020, as supplemented by letters dated March 17, 2021, April 1, 2021, and August 4, 2021, 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 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;

   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 the 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. 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. The license amendment is effective as of its date of issuance and shall be implemented no later than February 15, 2022.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

Digitally signed by James G. Danna
Date: 2021.10.08 13:46:59 -04'00'

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: October 8, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 339

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
I	I
XVIII	XVIII
1-5	1-5
3/4 1-10	3/4 1-10
3/4 4-3	3/4 4-3
3/4 4-3b	3/4 4-3b
3/4 4-24	3/4 4-24
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26
3/4 4-27	3/4 4-27
3/4 4-30	3/4 4-30
3/4 5-6	3/4 5-6
3/4 5-6a	3/4 5-6a
6-24b	6-24b
-	6-24c

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; and

(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.

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. 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) 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

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PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the Updated FSAR, 2) authorized under the provisions of 10CFR50.59, or 3) otherwise by the Commission.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.20a The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the Overpressure Protection System setpoint and enable temperature, for the current reactor vessel fluence period. The pressure and temperature limits shall be determined for each fluence period in accordance with Technical Specification Section 6.9.1.11.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.#

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

A maximum of one centrifugal charging pump shall be OPERABLE while in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, MODE 5, or MODE 6 when the head is on the reactor vessel.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (11) and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop (12) and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop (13) and its associated steam generator and reactor coolant pump,*
4. Reactor Coolant Loop (14) and its associated steam generator and reactor coolant pump,*
5. Residual Heat Removal Loop (11),
6. Residual Heat Removal Loop (12).

b. At least one of the above coolant loops shall be in operation.**

APPLICABILITY: MODE 4

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

---

* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to the POPS enable temperature specified in the PTLR unless 1) the pressurizer water volume is less than 1650 cubic feet (93.2% of pressurizer level indication) or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.4 Two# residual heat removal loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5.##

ACTION:

a. With less than the above required loops operable, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

One RHR loop may be inoperable for up to two hours for surveillance testing, provided the other RHR loop is OPERABLE and in operation. Additionally, four filled reactor coolant loops, with at least two steam generators with their secondary side water levels greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop.

A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to the POPS enable temperature specified in the PTLR unless 1) the pressurizer water volume is less than 1650 cubic feet (equivalent to approximately 93.2% of level), or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

* Systems supporting RHR loop operability may be excepted as follows:

  a. The normal or emergency power source may be inoperable.

** The residual heat removal pumps may be de-energized for up to 2 hours provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limits specified in the PTLR with:

a. A maximum heatup rate within the limits specified in the PTLR,

b. A maximum cooldown rate within the limits specified in the PTLR, and

c. A maximum temperature change within limits specified in the PTLR during hydrostatic testing operations above system design pressure.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits specified in the PTLR in accordance with the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update the P-T Limit Curves specified in the PTLR.

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LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two Pressurizer Overpressure Protection System relief valves (POPS) with a lift setting of less than or equal to the value specified in the PTLR, or

b. A reactor coolant system vent of greater than or equal to 3.14 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, except when the reactor vessel head is removed.

ACTION:

a. With one POPS inoperable in MODE 4 and the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, either restore the inoperable POPS to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

b. With one POPS inoperable in MODES 5 or 6 with the Reactor Vessel Head installed, restore the inoperable POPS to OPERABLE status within 24 hours, or complete depressurization and venting of the RCS through at least a 3.14 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

c. With both POPSs inoperable, depressurize and vent the RCS through a 3.14 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

d. In the event either the POPS or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the POPS or vent(s) on the transient and any corrective action necessary to prevent recurrence.

e. LCO 3.0.4.b is not applicable when entering MODE 4.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - Tavg < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump# and associated flow path capable of taking suction from the refueling water storage tank and transferring suction to the residual heat removal pump discharge piping and;

    1. Discharging into each Reactor Coolant System (RCS) cold leg.

b. One OPERABLE residual heat removal pump and associated residual heat removal heat exchanger and flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation and;

    1. Discharging into each RCS cold leg, and; upon manual initiation,

    2. Discharging into two RCS hot legs.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

d. LCO 3.0.4.b is not applicable to ECCS high head subsystem

A maximum of one safety injection pump or one centrifugal charging pump shall be OPERABLE in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, Mode 5, or Mode 6 when the head is on the reactor vessel.

ECCS SUBSYSTEMS - Tavg < 350°F

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All safety injection pumps and centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated to be inoperable in accordance with the Surveillance Frequency Control Program while in MODE 4 and the temperature of one or more of the RCS cold legs is less than or equal to the POPS enable temperature specified in the PTLR, MODE 5, or MODE 6 when the head is on the reactor vessel by either of the following methods:

a. By verifying that the motor circuit breakers have been removed from their electrical power supply circuits or,

b. For verifying that the pump is in a recirculation flow path and that two independent means of preventing RCS injection are utilized.

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results.

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG,

f. The results of any SG secondary side inspections;

g. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

h. The calculated accident induced leakage rate from the portion of the tubes below 15.21 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.16 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined,

i. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

6.9.1.11 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, POPS enable temperature, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.1.2.3, "Charging Pump-Shutdown"
2. Specification 3.4.1.3, “Reactor Coolant System Shutdown”
3. Specification 3.4.1.4, “Reactor Coolant System Cold Shutdown"
4. Specification 3/4.4.9.1, "RCS Pressure/Temperature Limits"
5. Specification 3.4.9.3, "Overpressure Protection Systems"
6. Specification 3/4.5.3, “ECCS Subsystems - Tavg < 350°F”

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC.

1. WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May, 2004
2. WCAP-18124-NP-A, Rev 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2019, may be used as an alternative to Section 2.2 of WCAP-14040-A Rev. 4.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplements thereto.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 DELETED

6.9.4 When a report is required by ACTION 1, 4, 8 or 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 339

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-70

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-272

1.0 INTRODUCTION

By letter dated December 6, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20343A128), as supplemented by letters dated March 17, 2021, April 1, 2021, and August 4, 2021 (ADAMS Accession Nos. ML21076A450, ML21091A246, ML21216A070, respectively), PSEG Nuclear LLC (the licensee) submitted a license amendment request (LAR) to modify technical specification (TS) Section 1.0, "Definitions," Section 3/4, "Limiting Conditions for Operation and Surveillance Requirements," and Section 6.0, "Administrative Controls," by replacing the existing reactor vessel heatup and cooldown rate limits and the pressure and temperature (P-T) limit curves with references to the pressure and temperature limits report (PTLR) at Salem Nuclear Generating Station (Salem), Unit No. 1. The LAR also proposed to update the existing P-T limits to extend their applicability through the period of extended operation to 47.4 effective full-power years (EFPY).

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90, the licensee proposed to revise the Salem, Unit No. 1, TSs to relocate the P-T limits for the reactor pressure vessel (RPV) to a licensee-controlled PTLR. Specifically, the licensee proposed to (1) modify TS Section 1.0, "Definitions," (2) delete the P-T curves in TS 3/4.4.9, "Pressure/Temperature Limits," (3) update references in TS Section 3/4 to the pressurizer overpressure protection system (POPS) setpoint to refer to the PTLR, and (4) modify TS Section 6.0, "Administrative Controls," to add 6.9.1.11 describing the approved analytical methods applied to evaluate P-T limit curves. The licensee provided the proposed PTLR in Enclosure 2 to the LAR, including the updated P-T limit curves for 47.4 EFPY.

The supplemental letters dated March 17, 2021, April 1, 2021, and August 4, 2021, 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, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 26, 2021 (86 FR 7117).

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.36, “Technical specifications,” state, in part, that each operating license will include TSs and that the TSs will include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulations in 10 CFR 50.36 also state that a summary statement of the bases for such TSs shall be included in applications for operating licenses, but shall not become part of the TSs.

The regulations in 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” require: (1) sufficient fracture toughness for RPV ferritic materials to provide adequate safety margins during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests; (2) that P-T limits satisfy the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G, and the minimum temperature requirements during normal heatup, cooldown, and pressure test operations; and (3) that applicable surveillance data from RPV material surveillance programs developed in accordance with 10 CFR Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” be incorporated into the calculations of P-T limits.

Generic Letter (GL) 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” advises licensees that they may request a license amendment to relocate the P-T limit curves from their plant TSs to a PTLR (ADAMS Accession No. ML031110004). GL 96-03 provides guidance that to do this licensees should: (1) generate their P-T limits in accordance with an NRC-approved methodology; (2) comply with 10 CFR Part 50, Appendices G and H; (3) reference NRC-approved methodologies in the TS; (4) define the PTLR in TS Section 1.0; (5) develop a PTLR to contain the P-T limit curves; and (6) modify applicable sections of the TSs accordingly.

Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” describes general procedures acceptable to the NRC staff for calculating the adjusted nil-ductility transition reference temperature RT_{NDT} (ART) due to neutron irradiation on RPVs (ADAMS Accession No. ML003740284).

Regulatory Issue Summary (RIS) 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” provides evaluation guidance for P-T limit curves and PTLRs, including the consideration of neutron fluence and structural discontinuities in the development of P-T limit curves (ADAMS Accession No. ML14149A165).

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (SRP), Section 5.3.2, “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock,” provides an acceptable method for determining the P-T limits based on the methodology of the ASME Code, Section XI, Appendix G (ADAMS Accession No. ML070380185).

RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” provides guidance regarding neutron fluence calculations (ADAMS Accession No. ML010890301).

By letter dated March 21, 2002 (ADAMS Accession No. ML020800488), the NRC approved the use of Technical Specifications Task Force (TSTF) Traveler TSTF-419-A, Revision 0, “Revise

PTLR Definition and References in ISTS [Improved Standard Technical Specifications] 5.6.6, RCS [Reactor Coolant System] PTLR" (ADAMS Accession No. ML012690234). By letter dated August 4, 2011 (ADAMS Accession No. ML110660285), the NRC staff clarified the use of TSTF-419-A.

The NRC approved topical report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves," as an acceptable method for developing P-T limit curves and POPS setpoints (ADAMS Accession No. ML050120209). The NRC imposed the following conditions on the use of WCAP-14040-A, Revision 4.

- Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements discussed in provision 2 in the table of Attachment 1 to GL 96-03 related to the RPV material surveillance program.

- Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision (Revision 4) of TR [topical report] WCAP-14040 should be modified to reflect this NRC staff conclusion.

- As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

3.0 TECHNICAL EVALUATION

3.1 Licensee Proposed Changes to TSs

The licensee proposed revisions to applicable sections of the TSs as shown below. The licensee stated that the proposed changes are consistent with the guidance in GL 96-03, as supplemented by TSTF-419-A.

(a) TS Section 1.0, "DEFINITIONS"

Add a new definition, "Pressure and Temperature Limits Report (PTLR)." The wording for this definition would be consistent with that in TSTF-419-A.

(b) TS Section 3/4, "LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS"

Delete the P-T curves and the associated TS wording and replace them with references to the PTLR.

Update references to the POPS setpoint to refer to the PTLR.

(c) TS Section 6.0, “ADMINISTRATIVE CONTROLS”

Add TS 6.9.1.11 describing the approved analytical methods applied to evaluate P-T limit curves.

3.2 Licensee Neutron Fluence Calculations

The current Salem, Unit No. 1, P-T limit curves expire at 32 EFPY, which Salem, Unit No. 1, is expected to reach by approximately February 15, 2022. The licensee proposed a set of revised P-T limit curves that were generated through the 60 years end of license extension (EOLE) (i.e., 50 EFPY), but would only be applicable through 47.4 EFPY to ensure compliance with 10 CFR 50.61, “Fracture toughness requirements for protection against pressurized thermal shock events.”

The Salem, Unit No. 1, EOLE neutron fluence projected at 50 EFPY is provided in WCAP-18502-NP, Revision 2, “Salem Unit 1 Heatup and Cooldown Limit Curves for Normal Operation” (Enclosure 3 of the LAR). The neutron fluence calculations were performed using the three-dimensional discrete ordinates code, RAPTOR-M3G, the BUGLE-96 cross-section library, and the least-squares evaluation FERRET Code for the surveillance capsule dosimetry that were approved and described in WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET” (Attachment 3 to Enclosure 1 of the LAR).

3.3 Licensee Proposed Pressure-Temperature Limits

The proposed Salem, Unit No. 1, PTLR is presented in Enclosure 2 of the LAR. The licensee stated that it prepared the P-T limit curves and PTLR in accordance with GL 96-03; ASME Code, Section XI, Appendix G; and WCAP-14040-A. Specifically, the licensee calculated: (1) the P-T limits in accordance with WCAP-14040-A, Revision 4, and ASME Code, Section XI, Appendix G and (2) the ART values for the limiting beltline materials in accordance with RG 1.99, Revision 2 (as discussed in GL 96-03).

The development of the 60-year Salem, Unit No. 1, P-T limit curves is documented in detail in Enclosure 3 of the LAR, WCAP-18502-NP. The licensee stated that all vessel materials subject to a fluence greater than 1×10^{17} n/cm^2 were considered in developing the P-T curves, consistent with RIS 2014-11. The licensee stated that the inlet and outlet nozzles were considered as part of the P-T limit curve evaluation due to the stress concentration at the nozzle corner. Table 3-1 of WCAP-18502-NP shows Copper and Nickel content of the vessel materials, as well as unirradiated RT_{NDT}.

Section 4 of WCAP-18502-NP describes the available data for calculation of Chemistry Factors (CF) according to RG 1.99, Revision 2, Regulatory Position 2.1. The licensee presented relevant sister plant surveillance data. Section 5 of WCAP-18502-NP describes the calculation of the CFs. Table 5-3 of WCAP-18502-NP shows the CFs calculated according to Regulatory Positions 1.1 and 2.1. ART values calculated according to RG 1.99, Revision 2, were reported by the licensee in Tables 7-2 and 7-3 of WCAP-18502-NP.

The licensee described the methodology for calculating P-T limit curves in Section 6 of WCAP-18502-NP. The licensee stated that the methodology is consistent with ASME Section XI, Nonmandatory Appendix G and WCAP-14040-A. The licensee reported the 47.4 EFPY P-T limit curves in Figures 8-1 and 8-2 of WCAP-18502-NP.

3.4. NRC Staff Evaluation

The NRC staff evaluated: (1) neutron fluence calculations in accordance with RG 1.190; (2) the proposed PTLR implementation in accordance with guidance in GL 96-03 and TSTF-419-A; and (3) the proposed P-T limit curves in accordance with WCAP-14040-A, 10 CFR Part 50, Appendices G and H, and the ASME Code, Section XI, Nonmandatory Appendix G.

3.4.1 Neutron Fluence Evaluation

The NRC staff reviewed the LAR and associated documentation referred to in the LAR for neutron fluence projections. Based on the regulatory evaluation in Section 2.0 of this safety evaluation (SE), the staff reviewed the neutron fluence projected at 60 years (50 EFPY) for the Salem, Unit No. 1, RPV materials that should be considered in the development of P-T limit curves due to a predicted neutron fluence exposure greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV). This review included the additional materials that are outside the traditional beltline region but that have the potential to exceed this fluence threshold, referred to as the “extended beltline” materials.

To apply the computer codes and cross-section library as described in WCAP-18124-NP-A to the calculation of neutron fluence, the licensee performed an applicability evaluation of WCAP-18124-NP-A and presented the results in LAR Section 3.2 for neutron fluence calculations.

The licensee concluded that WCAP-18124-NP-A is applicable to Salem, Unit No. 1. Specifically, the licensee applied the methodology described in WCAP-18124-NP-A to Salem, Unit No. 1, traditional and extended beltline (i.e., non-beltline) materials including reactor coolant system inlet and outlet nozzles. In performing the fast neutron exposure evaluations for Salem, Unit No. 1, RPV, the licensee conducted a series of fuel-cycle-specific forward transport calculations by using the three-dimensional discrete ordinates computer code RAPTOR-M3G, the BUGLE-96 cross-section library, and the least-squares evaluation FERRET code.

For Salem, Unit No. 1, transport calculations, the reactor model was constructed to include the necessary RPV details encompassing the traditional beltline region as well as the inclusion of the surveillance capsules and associated support structures (as described in Section 2.2 of WCAP-18502-NP). The NRC staff determined that the spatial mesh and angular quadrature and the pointwise inner iteration flux convergence criterion as utilized with this reactor model for WCAP-18502-NP are in conformance with RG 1.190 and are, therefore, acceptable.

Regarding the development of source distribution used in the transport calculations, the relevant information was outlined in Section 2.2 of WCAP-18502-NP. The NRC staff determined that the preparation of core neutron source for the transport calculations is in conformance with RG 1.190 and is, therefore, acceptable.

Therefore, the results from the neutron transport calculations provided data in terms of fuel cycle-averaged neutron flux, which when multiplied by the appropriate fuel cycle length, would generate the incremental fast neutron exposure for each fuel cycle until the 60 years EOLE.

Based on its review of WCAP-18502-NP, WCAP-14040-A, and associated references, the NRC staff determined that an evaluation of the dosimetry sensor sets from the surveillance capsules withdrawn from Salem, Unit No. 1, was provided. The dosimetry analyses documented showed that the ±20 percent (1σ) acceptance criterion specified in RG 1.190 is met.

In addition to the traditional beltline materials, the licensee also identified the RPV non-beltline materials and their locations in Tables 1 and 2 of the LAR, Enclosure 1, Attachment 3, for the development of P-T limits. Among them, the licensee explicitly considered the RPV materials with structural discontinuities, such as nozzles, as mentioned in RIS 2014-11.

The NRC staff determined from WCAP-18502-NP that the model used for the transport calculations as mentioned above had been expanded to axially and azimuthally encompass the non-beltline materials to calculate the neutron fluence to 50 EFPY. The licensee then performed uncertainty analysis to demonstrate whether the ±20 percent (1σ) acceptance criterion specified in RG 1.190 is met. Since the methodology used to generate WCAP-18502-NP (i.e., WCAP-18124-NP-A) is based on RG 1.190 and since RG 1.190 is guidance to project neutron fluence for the beltline region and not directly applicable to the extended beltline (i.e., non-beltline) region, the licensee further addressed, in the LAR, Enclosure 1, Attachment 3, the limitations and conditions of WCAP-18124-NP-A imposed on applying RAPTOR-M3G and FERRET to calculate the neutron fluence for the extended beltline region.

Based on the radiation transport calculation results and the traditional beltline and extended beltline (i.e., non-beltline) materials information, the NRC staff confirmed that the licensee had tabulated and transmitted the fast neutron (E > 1.0 MeV) fluence projections to 50 EFPY for both beltline and extended beltline materials from WCAP-18502-NP to the LAR for Salem, Unit No. 1. The staff concludes that the projection meets the acceptance criterion specified in 10 CFR Part 50, Appendix G, and conforms with the guidance in RIS 2014-11 because the licensee applied the approved methodology of WCAP-18124-NP-A to Salem, Unit No. 1, RPV materials that have the potential to experience a neutron fluence exposure greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at EOLE for the development of P-T limit curves.

3.4.2 PTLR Implementation

The NRC staff evaluated the proposed Salem, Unit No. 1, PTLR in accordance with the seven criteria in Attachment 1 to GL 96-03 as discussed below.

<u>Criterion 1</u>

Criterion 1 requires that the PTLR methodology describe the transport calculation methods including computer codes and formulas used to calculate neutron fluence values.

The proposed Salem, Unit No. 1, PTLR (Enclosure 2 to the LAR) describes the neutron fluence calculation methodology in Section 4.3. Therefore, the NRC staff finds that the licensee has satisfied Criterion 1 of GL 96-03.

<u>Criterion 2</u>

Criterion 2 requires that the PTLR methodology describe the reactor vessel material surveillance program.

The proposed Salem, Unit No. 1, PTLR (Enclosure 2 to the LAR) contains a description of the licensee's reactor vessel material surveillance program. The licensee stated that four capsules have been tested and referenced the corresponding capsule testing reports. The licensee stated that four capsules remain in the vessel and that one may be pulled for testing during the license renewal period. The NRC staff finds that the licensee has satisfied Criterion 2 of

GL 96-03 because the licensee described the 10 CFR Part 50, Appendix H, program in Appendix A of the proposed PTLR.

Criterion 3

Criterion 3 requires that the PTLR methodology describe how the low temperature overpressure protection system limits are calculated applying system/thermal hydraulics and fracture mechanics.

The licensee described the POPS (i.e., low temperature overpressure protection system) in Section 4.14 of the proposed Salem, Unit No. 1, PTLR (Enclosure 2 to the LAR). The licensee referenced the NRC-approved POPs methodology and described the existing POPs setpoint of 375 psig. The licensee stated that the new setpoint as analyzed by the approved methodology is 429 psig. Given the use of the approved methodology, the staff finds that the licensee may update the PTLR to the new setpoint without submission to the NRC for approval, provided that the NRC approves the use of the proposed PTLR in this SE.

The NRC staff finds that the licensee has satisfied Criterion 3 of GL 96-03 because of the discussion included in Section 4.14 of the proposed PTLR.

Criterion 4

Criterion 4 requires that the PTLR methodology describe the method for calculating the ART values using RG 1.99, Revision 2, and that the licensee identify the RT_{PTS} value in accordance with 10 CFR 50.61.

The licensee described the calculation of the ART values in Section 4.10 of the proposed Salem, Unit No. 1, PTLR (Enclosure 2 to the LAR). The licensee referenced WCAP-18502-NP (Enclosure 3 to the LAR) as describing the methodology to calculate ART. The NRC staff verified that WCAP-18502-NP implements the RG 1.99, Revision 2, methodology for calculating ART. Further, the staff performed confirmatory calculations for the licensee's ART values given in Table 7-2 of WCAP-18502-NP. The staff's independent calculations confirmed these values, providing reasonable assurance that the licensee correctly implemented the guidance of RG 1.99, Revision 2. The licensee provided the limiting ART values in Table 4-4 of the proposed PTLR.

The licensee described the calculation of RT_{PTS} in Section 4.11 of the proposed Salem, Unit No. 1, PTLR (Enclosure 2 to the LAR). The licensee referenced WCAP-18502-NP, Tables G-1 and G-2. The licensee stated that Lower Shell Longitudinal Weld Seam 3-042C exceeded the 10 CFR 50.61 pressurized thermal shock screening criteria of 270 °F at 50 EFPY. The licensee stated that the RT_{PTS} values were recalculated to 47.4 EFPY to maintain compliance with the 10 CFR 50.61 screening limit. The NRC previously approved this calculation as part of reactor vessel neutron embrittlement time-limited aging analyses in Section 4.2.3 of the Safety Evaluation Report related to the license renewal of Salem, Unit No. 1 (ADAMS Accession No. ML11166A135).

The NRC staff finds that the licensee has satisfied Criterion 4 of GL 96-03 because of the discussion included in Sections 4.10 and 4.11 of the proposed PTLR.

Criterion 5

Criterion 5 requires that the PTLR methodology describe the application of fracture mechanics in the construction of P-T limits based on the ASME Code, Section XI, Appendix G, and SRP Section 5.3.2.

The licensee described the proposed operating limits, including the P-T limits, in Section 3 of the proposed Salem, Unit No. 1, PTLR (Enclosure 2 to the LAR). The license provided the underlying technical background of the operating limits in Section 4 of the proposed PTLR. The licensee stated that the NRC-approved methodology is WCAP-14040-A, Revision 4. The NRC staff performed confirmatory calculations on the proposed EFPY 47.4 P-T limits and found that the proposed curves were reasonable.

The NRC staff finds that the licensee has satisfied Criterion 5 of GL 96-03 because of the discussion included in Sections 3 and 4 of the proposed PTLR.

Criterion 6

Criterion 6 requires that the PTLR methodology describe how the minimum temperature requirements such as minimum bolt-up temperature and hydrotest temperature in Appendix G to 10 CFR Part 50 are applied to P-T curves.

The licensee provided the minimum bolt-up temperature and hydrotest temperature in Figures 3-1 and 3-2 of the proposed Salem, Unit No. 1, PTLR (Enclosure 2 to the LAR). The methodology for calculating these values is described in WCAP-18502-NP (Enclosure 3 to the LAR) and NRC-approved WCAP-14040-A, Revision 4.

The NRC staff finds that the licensee has satisfied Criterion 6 of GL 96-03 because of the annotations of Figures 3-1 and 3-2 in the proposed PTLR.

Criterion 7

Criterion 7 requires that the PTLR methodology describe how the data from multiple surveillance capsules are used in the ART calculation.

The licensee described the use of surveillance data in Sections 4.5 and 4.6 of the proposed Salem, Unit No. 1, PTLR (Enclosure 2 to the LAR), along with the data credibility evaluation in Appendix C of the proposed PTLR. The licensee stated that capsule data from the Salem, Unit No. 1, plant-specific program and sister plant capsule data were considered in the calculation of RG 1.99, Revision 2, Regulatory Position 2.1 chemistry factors. The licensee stated that Regulatory Position 1.1 chemistry factors bound those of Regulatory Position 2.1.

The NRC staff finds that the licensee has satisfied Criterion 7 of GL 96-03 because of the discussion in Sections 4.5 and 4.6 of the proposed PTLR.

Conclusion

Based on the above, the NRC staff concludes that the licensee has satisfied all seven of the criteria of Attachment 1 to GL 96-03.

3.4.3 P-T Limits and POPS Evaluation

The NRC staff performed confirmatory calculations of the licensee's proposed P-T limit curves for 47.4 EFPY and found the proposed curves to be reasonable. Furthermore, the licensee implemented the NRC-approved methodology of WCAP-14040-A to calculate the proposed P-T limit curves. Therefore, the staff finds that the proposed P-T limit curves for 47.4 EFPY are acceptable.

The NRC staff reviewed the LAR and the Westinghouse (vendor of analysis) LTR-SCS-20-28 referred to in the LAR for POPS analysis. The staff found that the application of the NRC-approved methodology, WCAP-14040-A, by the vendor to perform the POPS analysis for Salem, Unit No. 1, is acceptable because, as stated in the SE of WCAP-14040-A, Salem, Unit No. 1, is a Westinghouse-designed pressurized water reactor to the extent specified and under the limitations via technical requirements delineated in the report. As described in the SE of WCAP-14040-A, the vendor should perform the POPS analysis as one of the necessary topics required to address the 10 CFR Part 50, Appendix G, regulatory requirements.

For the POPS analysis, the SE of WCAP-14040-A specifies in the 7th technical requirement that the lift setting limits for the power operated relief valves (PORVs) should be developed using NRC-approved methodologies. It further states that the thermal hydraulics analysis for the mass and heat input transients should use the same specialized version of LOFTRAN, which was approved in WCAP-14040, Revision 2. The NRC staff confirmed from the licensee's request for additional information response (ADAMS Accession No. ML21091A246) that the vendor had used the equivalently specialized version of LOFTRAN, as approved in WCAP-14040, Revision 2, for the low-temperature overpressure protection (LTOP) analysis.

The NRC staff's findings from its review of the LAR and Westinghouse LTR-SCS-20-28 are summarized below:

a. The licensee transmitted the Salem, Unit No. 1, key input parameters to the vendor of analysis and confirmed with the vendor of analysis that the bounding values and conservative analysis assumptions were used.

b. The vendor of analysis used the mass injection (MI) and heat injection (HI) transient analysis results, steady-state (isothermal) Appendix G P-T limits for Salem, Unit No. 1, valid up to 47.4 EFPY, and the licensee-provided pressure and temperature uncertainties to develop the POPS PORV setpoints in accordance with WCAP-14040 methodology.

c. The vendor of analysis determined the final maximum allowable PORV setpoint such that it bounds both the MI and HI transient maximum allowable PORV setpoints for RCS temperature up to 325°F, which was used to conservatively determine the POPS enable temperature. The licensee will specify both PORV setpoint and POPS enable temperature in the PTLR.

With the above evaluation results, the NRC staff finds that the LTOP (or POPS) analysis for Salem, Unit No. 1, as well as the POPV lifting setpoint, is acceptable for Salem, Unit No. 1, to support operation up to 47.4 EFPY because the vendor applied the approved methodology of WCAP-14040 to Salem, Unit No. 1, LTOP analysis for the licensee to meet the 10 CFR Part 50, Appendix G regulatory requirement of protecting the RPV from brittle failure with conservative overpressure mitigating system setpoints and POPS enable temperature.

3.5 Technical Conclusion

Based on the information submitted, the NRC staff determined that: (1) the proposed P-T limit curves in the Salem, Unit No. 1, PTLR considered all RPV materials that have the potential to experience a neutron fluence exposure greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at EOLE; (2) the proposed PTLR was implemented in accordance with guidance in GL 96-03 and TSTF-419-A; and (3) the proposed P-T limit curves were determined based on the NRC-approved methodology in WCAP-14040-A, 10 CFR Part 50, Appendices G and H, and the ASME Code, Section XI, Nonmandatory Appendix G.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment on April 5, 2021. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The NRC has previously issued a proposed finding that the amendment involves no significant hazards consideration, as published in the *Federal Register* (86 FR 7117; January 26, 2021), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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