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

February 23, 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 NOS. 1 AND 2 − ISSUANCE OF AMENDMENT NOS. 336 AND 317 RE: LEAK-BEFORE-BREAK FOR ACCUMULATOR, RESIDUAL HEAT REMOVAL, SAFETY INJECTION, AND PRESSURIZER SURGE LINES (EPID L-2020-LLA-0088)

Dear Mr. Carr:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 336 and 317 to Renewed Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2, respectively. These amendments consist of changes to the Salem, Unit Nos. 1 and 2, Updated Final Safety Analysis Report in response to your application dated April 24, 2020, as supplemented by letter dated September 16, 2020.

The amendments made changes to the Salem Updated Final Safety Analysis Report to use the leak-before-break methodology to eliminate the dynamic effects of postulated pipe ruptures in specific portions of systems attached to the reactor coolant system.

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 Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 336 to DPR-70
2. Amendment No. 317 to DPR-75
3. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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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. 336
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 Exelon Generation Company, LLC (the licensees), dated April 24, 2020, as supplemented by letter dated September 16, 2020, 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, by Amendment No. 336, Renewed Facility Operating License No. DPR-70 is hereby amended to authorize the change to the Updated Final Safety Analysis Report (UFSAR) as requested by letter dated April 24, 2020, as supplemented by letter dated September 16, 2020, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance. The licensee shall submit the update of the UFSAR authorized by this amendment in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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

Date of Issuance: February 23, 2021

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

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 317
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 Exelon Generation Company, LLC (the licensees), dated April 24, 2020, as supplemented by letter dated September 16, 2020, 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, by Amendment No. 317, Renewed Facility Operating License No. DPR-75 is hereby amended to authorize the change to the Updated Final Safety Analysis Report (UFSAR) as requested by letter dated April 24, 2020, as supplemented by letter dated September 16, 2020, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance. The licensee shall submit the update of the UFSAR authorized by this amendment in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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

Date of Issuance: February 23, 2021

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 336 AND 317 TO

RENEWED FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated April 24, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20115E374), as supplemented by letter dated September 16, 2020 (ADAMS Accession No. ML20260H194), PSEG Nuclear LLC (PSEG, the licensee) submitted a license amendment request (LAR) for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. The proposed amendments would apply the leak-before-break (LBB) methodology to specific portions of emergency core cooling system (ECCS) accumulator, residual heat removal (RHR), and safety injection (SI) lines and entire pressurizer surge lines. Enclosures 8 through 12 to the application contain sensitive unclassified non-safeguards information and, per the licensee's request, have been withheld from public disclosure pursuant to Section 2.390, "Public inspections, exemptions, requests for withholding," of Title 10 of the *Code of Federal Regulations* (10 CFR).

The request is intended to eliminate the dynamic effects of postulated pipe ruptures in the subject piping from the design basis of Salem, Unit Nos. 1 and 2, in accordance with General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The request does not propose changes to the technical specifications (TSs).

The supplemental letter dated September 16, 2020, 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 July 7, 2020 (85 FR 40692).

2.0 REGULATORY EVALUATION

2.1 Description of System Design and Operation

Salem, Unit Nos. 1 and 2, are four-loop Westinghouse pressurized-water reactor (PWR) plants. The Salem ECCS design includes the flow and piping configurations as follows: (a) passive injection via four ECCS accumulators, (b) low-head injection via two RHR pumps (also used for decay heat removal during shutdown conditions), (c) high-head SI via two centrifugal charging pumps (also used for normal charging flow during plant operation), and (d) intermediate-head injection via two SI pumps. As described in the Salem, Unit Nos. 1 and 2, Updated Final Safety Analysis Report (UFSAR), Section 3.6.4.2.1, the current design basis includes application of LBB to the large reactor coolant system (RCS) primary loop piping.

The LBB concept is based on calculations and experimental data demonstrating that certain pipe materials have sufficient fracture toughness to prevent a small through-wall crack from propagating rapidly and unstably to a catastrophic pipe rupture and to ensure that the probability of a pipe rupture is extremely low. LBB is used to demonstrate that small through-wall cracks in the pipe are stable, and the associated leakage will be detected by the RCS leakage detection systems promptly so that operators can shut down the reactor and take corrective actions before the pipe ruptures.

2.2 Description of Licensee Proposed Change

This LAR would expand the scope of the LBB methodology to include specific portions of piping systems attached to the primary loop piping. These portions include ECCS high-head SI, intermediate head injection, low-head injection, passive injection via four ECCS accumulators, and pressurizer surge lines. The expanded scope of LBB lines addressed in this application have a smaller piping size than the previously approved LBB applications of the licensee. The expanded scope LBB would eliminate the dynamic effects of postulated ruptures of specific portions of piping for SI, RHR, ECCS accumulators, and the pressurizer surge line. Approval of this request to expand the scope of LBB would affect information in the following sections of the Salem UFSAR: (a) Section 3.6.1, “Systems in Which Design Basis Piping Breaks Occur”; (b) Section 3.6.4.2.1, “Postulated Break Locations”; (c) Section 3.6.6, “References for Section 3.6”; (d) Table 3.6-1, “Postulated Reactor Coolant System Pipe Ruptures,” and Figure 3.6-1 “Postulated Break Locations – Reactor Coolant System”; (e) Section 5.2.1.7, “Protection against the Proliferation of Dynamic Effects,” and Section 5.2.9, “References for Section 5.2”; (f) Section 5.2.7, “Reactor Coolant Pressure Boundary Leakage Detection Systems”; and (g) Appendix B, Section A.4.4.3, “Leak-Before-Break Analyses.”

The LBB evaluations rely on the licensee’s ability to detect unidentified RCS leakage and take the appropriate actions prior to the TS limit to preclude pipe rupture.

2.3 Current Technical Specification Requirements

Section 2.2, “Current Technical Specification Requirements,” of Enclosure 1 to the LAR states that the current Salem TSs associated with this LAR are Unit No. 1 TS 3/4.4.6 and Unit No. 2 TS 3/4.4.7, “Reactor Coolant System Leakage.”

Salem Unit No. 1 TS 3/4.4.6.1 and Unit No. 2 TS 3/4.4.7.1 each require three RCS leakage detection systems to be operable during plant operation in Modes 1, 2, 3, and 4:

- containment atmosphere particulate radioactivity monitoring system,
- containment sump level monitoring system, and
- either containment fan cooler condensate flow rate or containment gaseous radioactivity monitoring system.

The required actions for inoperable leakage detection systems are the same for both Salem units, as follows:

> With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TS surveillance requirements demonstrate operability of the leakage detection systems. The containment atmosphere particulate and gaseous monitors are demonstrated operable by channel check, source check, channel calibration, and channel functional test. Containment sump level and containment fan cooler flow rate measurement systems are demonstrated operable by channel calibration.

Unit No. 1 TS 3/4.4.6.2 and Unit No. 2 TS 3/4.4.7.2 each include limitations on RCS leakage during plant operation in Modes 1, 2, 3, and 4, including zero allowable pressure boundary leakage and 1 gallon per minute (gpm) limit for unidentified leakage. In accordance with the Surveillance Frequency Control Program, leakage is demonstrated to be within limits by surveillances, including 12-hour log readings of the containment atmosphere particulate monitor and the containment sump inventory, and by performing a water inventory balance at a 72-hour frequency. Current Salem operations procedures direct the performance of the water inventory balance every 24 hours.

Additionally, in Section 4.0, “Regulatory Evaluation,” of Enclosure 1 to the LAR, the licensee indicated, in part, that Criterion 1 of 10 CFR 50.36(c)(2)(ii) applies to the TS-required RCS leakage detection systems, which are used to detect degradation of the RCPB, and that Criterion 2 10 CFR 50.36(c)(2)(ii) applies to the TS limits on RCS operational leakage, including the unidentified leakage limit of 1 gpm. The licensee concluded that the existing TS for leakage detection and leakage limits are consistent with 10 CFR 50.36 and do not require revision to support this request. Therefore, there are no proposed changes to the Salem TSs associated with this proposed LBB change, as discussed in Section 3.4 of this safety evaluation.

2.4 Regulatory Requirements and Guidance

In accordance with 10 CFR 50.90, the licensee requested to amend its license to apply the LBB methodology to specific portions of ECCS accumulator, RHR, and SI lines and entire pressurizer surge lines.

GDC 4 in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 is the general design criteria that is directly applicable to the LAR.

GDC 4, “Environmental and dynamic effects design bases,” states, in part, that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. The current compliance of Salem Unit Nos. 1 and 2 with GDC 4 is described in UFSAR Section 3.1.3.

To meet GDC 4, a licensee needs to submit, for NRC staff review and approval, a fracture mechanics evaluation of specific piping configurations. The candidate piping should also satisfy the screening criteria of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition” (SRP), Section 3.6.3, “Leak-Before-Break Evaluation Procedures,” Revision 1, March 2007 (ADAMS Accession No. ML063600396), by demonstrating that it does not experience active degradation. The licensee should also demonstrate that the fracture mechanics analysis of the candidate piping meets the safety margins in SRP Section 3.6.3, Revision 1. In addition, the RCS leakage detection systems should be able to detect a certain leak rate with margins, compared to the leak rate from the leakage crack size of the subject piping in accordance with SRP Section 3.6.3, Revision 1.

The regulation at 10 CFR 50.36(b) requires TSs to be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto.

As required by 10 CFR 50.36(c)(2)(i), the TSs will include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor 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)(2)(ii)(A) states that a TS LCO must be established for:

> Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The regulation in 10 CFR 50.36(c)(2)(ii)(B) states that a TS LCO must be established for:

> Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 1 applies to the TS-required RCS leakage detection systems, which are used to detect degradation of the RCPB. Criterion 2 applies to the TS limits on RCS operational leakage, including the unidentified leakage limit of 1 gpm.

The regulation in 10 CFR 50.36(c)(3) requires TSs to include items in the category of surveillance requirements (SRs), which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition" (SRP), Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Revision 1, provides guidance on screening criteria, safety margins, and analytical methods for the piping systems to be qualified for the application of LBB.

NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," November 1984 (ADAMS Accession No. ML093170485), provides the technical basis for the LBB analysis.

Regulatory Guide (RG) 1.45, Revision 1, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," May 2008 (ADAMS Accession No. ML073200271), describes the acceptance criteria for the RCS leakage detection systems.

3.0 TECHNICAL EVALUATION

The NRC staff followed the guidance in SRP 3.6.3, Revision 1, to review the licensee's LBB analysis. The NRC staff reviewed the segments of piping lines for the LBB analysis (i.e., scope of LBB analysis), as discussed in Section 3.1 below. The NRC staff also evaluated whether the subject piping lines satisfy the screening criteria for various degradation mechanisms, as documented in Section 3.2 of this safety evaluation. The NRC staff further reviewed the limit load analysis of the subject piping, as discussed in Section 3.3. In addition, the NRC staff evaluated the capability of the RCS leakage detection systems, as addressed in Section 3.4.

As part of the submittal, the licensee provided the following Westinghouse non-proprietary reports that describe the LBB analysis (ADAMS Accession No. ML20115E374):

- WCAP-18248-NP, Revision 0, "Technical Justification for Eliminating Safety Injection Line Rupture as the Structural Design Basis for Salem Units 1 and 2, Using Leak-Before-Break Methodology"

- WCAP-18249-NP, Revision 0, "Technical Justification for Eliminating Accumulator Line Rupture as the Structural Design Basis for Salem Units 1 and 2, Using Leak-Before-Break Methodology"

- WCAP-18253-NP, Revision 0, "Technical Justification for Eliminating Residual Heat Removal Line Rupture as the Structural Design Basis for Salem Units 1 and 2, Using Leak-Before-Break Methodology"

- WCAP-18261-NP, Revision 0, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Salem Units 1 and 2, Using Leak-Before-Break Methodology"

- WCAP-18516-NP, Revision 1, "Fatigue Crack Growth Evaluations of Salem Units 1 and 2 Accumulator, RHR, Pressurizer Surge and Safety Injection Lines Supporting Expanded Scope Leak-Before-Break"

3.1 Scope of the LBB Analysis

The licensee's LBB analysis in the LAR pertains to specific portions of the accumulator, RHR, and SI lines that are connected to the RCS primary loops. The scope of the LBB analysis for the accumulator, RHR, and SI lines is limited to each piping segment from the primary loop up to the first RCS pressure isolation valve (PIV). In addition, the entire pressurizer surge line is included in the LBB analysis. Detailed piping configurations are presented in Chapter 3 of the WCAP-18248, 18249, 18253, and 18261 reports. The specific piping segments within the LAR scope are also described in Enclosure 1 of the LAR, dated April 24, 2020. The licensee analyzed these piping segments to credit the proposed LBB methodology at Salem, Unit Nos. 1 and 2. The following systems and associated piping segments were evaluated for NRC approval of the proposed LBB application.

3.1.1 Accumulator Lines

Each of the four cold-leg pipes is provided with an ECCS accumulator via a 10-inch line. The accumulators are designed to inject borated water into the reactor vessel if RCS pressure is reduced below the accumulator pressure as a result of a plant transient. During normal plant operation, each accumulator is isolated from the RCS by two check valves (SJ56 and SJ55 valves). For each cold leg, the scope of the LBB analysis is the SI piping starting at the cold leg up to the first check valve (SJ56 valve). The accumulator piping segment in the scope of the LAR is designated as the ACC-I segment.

The material for fabrication of the subject accumulator piping is A376 TP316 stainless steel and A403 WP316 stainless steel. The accumulator line nozzle at the RCS piping was fabricated with A182 F316 stainless steel. The licensee explained that the tensile properties of A376 TP316 (piping) and A182 F316 (nozzle) stainless steels are identical according to the material properties in the American Society of Mechanical Engineers (ASME) Code, Section II, "Materials." The licensee also clarified that the LBB stresses in the nozzle are significantly lower than the stresses in the piping due to a larger geometry for the distribution of stress. Based on these material properties and stress levels, the nozzle locations are bounded by the piping locations in the LBB analysis. Similarly, the licensee confirmed that the piping locations are bounding for the nozzle locations in the LBB analysis of RHR, SI, and pressurizer surge lines. In addition, the weld processes used in the fabrication of the accumulator piping are either shielded metal arc weld (SMAW) or submerged arc weld (SAW) process.

3.1.2 RHR Lines

At each unit, the 14-inch RHR pump suction line starts from the loop 1 hot leg and includes two motor-operated isolation valves in series (RH1 and RH2 valves). The scope of the LBB analysis for the RHR line at each unit is the piping segment from the loop 1 hot leg up to the first PIV. The piping segment in the scope of LAR is designated as the RHR-I segment.

The RHR return to the RCS cold legs is through the 6-inch SI lines that are attached to the 10-inch accumulator lines. For this return path, the scope of the LAR is from the RCS cold leg to the SJ56 check valve (ACC-I segment). The RHR return to the RCS hot legs is through the 6-inch SI lines that connect to the hot legs of loops 3 and 4. For this return path, the scope of the LAR is from the RCS hot leg to the SJ156 check valve, which is the first check valve off the RCS primary loop (SI-HL-I segment).

The material for fabrication of the subject RHR piping is A376 TP316 stainless steel and A403 WP316 stainless steel. The nozzle of this piping line at the RCS piping was fabricated with A182 F316 stainless steel. The welding process used in the piping fabrication is the SMAW or SAW process.

3.1.3 SI Lines

The licensee stated that each of the RCS cold legs is provided with a low-head SI flow path via a 6-inch line that connects to the 10-inch ECCS accumulator line, as shown in Figure 3-1 of the WCAP-18248 report. The connecting accumulator segment (segment ACC-II) is outside the first isolation valve (SJ56 valve) of the accumulator injection line and is excluded from the scope of the LAR. Therefore, the 6-inch SI line is isolated from each cold leg of the RCS during normal operation by the PIV. As identified above, the piping segment outside the PIV (segment SI-CL-I) is also excluded from the scope of this LAR.

The licensee further stated that the intermediate-head SI flow paths are connected to the hot legs and consist of three different configurations, as shown in Figures 3-2, 3-3, and 3-4 of the WCAP-18248 report. The segments that are in the scope of this LAR are represented by segment SI-HL-I. For loop 1, this segment in the LAR scope is from the 14-inch RHR line to the first PIV of the SI line. For the other loops, segment SI-HL-I is from the hot leg to the respective first PIV. These hot-leg SI piping segments are included in the scope of the LAR. The other segments of the hot-leg SI piping are isolated from the RCS during normal operation and are, therefore, excluded from the LAR scope.

The material for fabrication of the subject piping is A376 TP316 stainless steel and A403 WP316 stainless steel. The nozzle of this piping line at the RCS piping was fabricated with A182 F316 stainless steel. The welding process used in the piping fabrication is the SMAW or SAW process.

3.1.4 Pressurizer Surge Lines

The pressurizer surge line connects the bottom of the pressurizer to the loop 3 hot leg at each unit. Figures 3-1 and 3-2 of the WCAP-18261 report show the pressurizer surge line layout for Salem, Units Nos. 1 and 2, respectively. The scope of the LBB analysis in the LAR includes the entire surge line from the connection to the RCS hot-leg nozzle to the pressurizer nozzle connection. The pressurizer surge line is 14-inch (schedule 140) piping for Unit No. 1 and 14-inch (schedule 160) piping for Unit No. 2. The maximum normal operating temperature is approximately 650 degrees Fahrenheit (°F).

The material for fabrication of the subject piping is A376 TP316 stainless steel. The surge line nozzle at the RCS piping was fabricated with A182 F316 stainless steel. The subject piping is fabricated entirely of straight pipes and pipe bends and does not include an elbow or other pipe fitting. The welding process used in the piping fabrication is the SMAW or SAW process.

3.1.5 Summary of Analysis Scope

The NRC staff finds that the LAR, as supplemented, has clearly identified the specific portions of the accumulator, RHR, SI and pressurizer lines that are subject to this LBB analysis and, therefore, the scope of the LBB analysis is identified appropriately. The NRC staff concludes that the licensee may credit this LBB analysis for the piping segments that are within the scope

of this evaluation (i.e., as bounded by the piping segments identified in Sections 3.1.1 through 3.1.4 above).

3.2 Screening Based on Applicable Degradation Mechanisms

SRP Section 3.6.3.III specifies that active degradation should not be a potential source to cause pipe rupture in the application of LBB (e.g., degradation due to stress corrosion cracking (SCC), fatigue, water hammer, corrosion, wall thinning, creep, or brittle cleavage-type failure).

The NRC staff evaluated the LBB analysis in accordance with the degradation screening criteria of SRP Section 3.6.3.III, as discussed in the following sections.

3.2.1 Stress Corrosion Cracking

As discussed in Section 2.1 of WCAP-18248, the following elements or contaminants of a reactor coolant environment are known to increase the susceptibility of austenitic stainless steel to SCC: oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (e.g., sulfides and sulfites). During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to SCC. The water chemistry control standards are also included in the plant operating procedures. Therefore, during plant operation, the likelihood of SCC is minimized.

The Westinghouse RCS primary loops and connected Class 1 piping have an operating history that demonstrates the inherent operating stability characteristics of the design. The operating experience also confirms that the RCS piping is resistant to intergranular stress corrosion cracking. In comparison, the operating experience has shown that primary water stress corrosion cracking (PWSCC) has occurred in nickel-based Alloy 82/182 dissimilar metal butt welds in PWR coolant environment. The licensee confirmed that these materials, which are susceptible to PWSCC, are not used in the accumulator, RHR, SI, and pressurizer surge lines of Salem, Unit Nos. 1 and 2.

In addition, the licensee evaluated the susceptibility of the subject piping to external chloride-induced stress corrosion cracking (ECSCC) because Salem, Units Nos. 1 and 2, are coastal site plants, and brackish, chlorinated Delaware River water is used to provide containment building cooling via the service water system to the containment fan cooler units (CFCUs).

In its evaluation, the licensee concluded that the piping in the scope of the LAR is not considered to be susceptible to ECSCC due to its location relative to potential sources of service water leaks and spills. The LBB boundaries are located within the polar crane wall, which is a 3-foot thick concrete ring wall that surrounds RCS piping and associated components. In contrast, the areas potentially susceptible to chloride contamination and ECSCC are along the leakage path from the CFCU service water system piping. The licensee found that the piping in the scope of the LAR is not susceptible to service water exposure based on its location within the polar crane wall, away from the service water leakage paths. The licensee also evaluated Salem's risk-informed inservice inspection program and its inspection results related to ECSCC. The licensee confirmed that no relevant indication of ECSCC was found in the subject piping within the scope of the LAR.

The NRC staff notes that, as the operating experience demonstrates, PWSCC has been the prevailing active degradation mechanism in PWR Class 1 piping lines when fabricated with Alloy 82/182 dissimilar metal welds. However, the licensee confirmed that the subject piping lines do not contain Alloy 82/182 materials that are susceptible to PWSCC. Therefore, the NRC staff finds that SCC is not an active degradation mechanism for the subject piping based on the discussion above. In addition, the NRC staff finds that the potential for ECSCC in the subject piping is not a concern, based on the piping location away from the potential leakage paths involving chloride contamination and the ongoing inspection activities under the risk-informed inservice inspection program.

3.2.2 Fatigue

The licensee evaluated the piping susceptibility to low cycle and high cycle fatigue and the potential impact of fatigue on the piping integrity. In addition, the licensee provided the evaluation of thermal stratification that could cause fatigue. The NRC staff's evaluation on these matters is described below.

3.2.2.1 Low Cycle and High Cycle Fatigue

The licensee stated that the subject piping was originally designed in accordance with the United States of America Standards (USAS) B31.1 – 1967 Piping Code. The licensee also explained that piping designed in accordance with the B31.1 code is not required to have an analysis of cumulative fatigue usage, but cyclic loading is considered in a simplified manner in the design process. Instead of an explicit fatigue analysis, the subject piping complies with the provision that an adequate stress range reduction factor be applied to the allowable stress as needed to address fatigue effect from full temperature cycles for thermal expansion stress evaluation. The LAR, as supplemented, identified that the stress range reduction factor is 1.0 (i.e., no reduction) for equivalent full temperature cycles less than 7,000. For the subject piping, the equivalent full temperature cycles for the applicable design transients are less than 7,000, and therefore, no reduction is required for the stress range.

The licensee also indicated that Salem, Unit Nos. 1 and 2, are operating in the extended period of operation (i.e., operation beyond 40 years up to 60 years). As part of the Fatigue Monitoring Program for license renewal, the licensee performs an annual review of transient cycle counts and compares the transient data to the 7,000 cycle limit to confirm the fatigue cycle criteria are met, as discussed above.

The licensee also performed a fatigue time-limited aging analysis (TLAA) using cumulative usage factors (CUFs) as part of its license renewal. The analysis includes the evaluation of the effects of the reactor water environment on leading fatigue locations (highest design CUF locations) applicable to older vintage Westinghouse plants, consistent with the guidance in Section 5.5 of NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (ADAMS Accession No. ML031480219). This fatigue analysis was conducted in accordance with ASME Section III, Class 1 rules for each of the leading fatigue locations. In addition, environmentally assisted fatigue correction factors were calculated and applied to the CUF values developed in the Class 1 fatigue analysis to determine the environmental CUF for license renewal.

The fatigue TLAA also includes the pressurizer surge line nozzles at the RCS hot leg and the pressurizer. These nozzles are in the scope of the LAR. The calculated 60-year environmental CUF values for the leading fatigue locations (limiting locations) at Salem, Unit Nos. 1 and 2, are

less than the limit (i.e., 1.0) and are, therefore, acceptable. During the extended period of operation, the environmental CUF values at the leading locations are continuously tracked by the Fatigue Monitoring Program using the Westinghouse WESTEMS software to confirm that the environmental CUF limit is met.

The NRC staff noted that the design basis of Salem, Unit Nos. 1 and 2 relies on the B31.1 code provisions for fatigue analyses and the 60-year environmentally assisted fatigue TLAA for license renewal, which project environmental CUF values for the leading locations of the RCS piping. The NRC staff finds that these fatigue analyses support the evaluation results that degradation due to fatigue will not affect the structural integrity of the subject piping within this LAR's scope, and potential cracking due to fatigue will be adequately managed for the extended period of operation (60 years).

In addition, Section 8.0 of the WCAP-18248 report, which was submitted with the LAR, addresses the fatigue crack growth (FCG) analysis of a postulated circumferential inner-surface crack for the subject SI piping. This FCG analysis addresses crack growth of postulated initial flaws. The FCG analysis is compared with the environmental CUF analysis discussed above in that the CUF analysis focuses on the susceptibility to fatigue initiation.

Section 8.0 of WCAP-18248 indicates that the FCG analysis uses the representative operating and loading conditions for typical PWR plants, which bound those for the subject SI piping. WCAP-18248, Section 8.0, further states that only a limited number of transient cycles is expected to contribute to the growth of a postulated flaw between the time when leakage reaches 5 gpm (i.e., 0.5 gpm detection capability times a margin of 10) and the time that the reactor would be shut down due to the leakage. The WCAP report clarifies that, because a leakage detection limit (0.5 gpm) for the SI piping is less than the 1.0 gpm basis identified in RG 1.45, Revision 1, supplemental assessment of the potential for FCG has been conducted for the SI piping as an additional defense-in-depth justification.

WCAP-18248 presents the results of the representative FCG analysis that are bounding for the 6-inch SI piping in two formats, as follows: (1) maximum flaw growth over a 10-year period (related to inservice inspection (ISI) intervals), and (2) flaw growth over the total plant operating time (conservatively assumed as 80 years). In the FCG analysis for the 10-year period, a circumferential flaw with an initial depth of 0.309 inches is estimated to grow to a depth of 0.343 inches. In the FCG analysis for the 80 years of operation, a circumferential flaw with an initial depth of 0.067 inches is estimated to grow to a depth of 0.128 inches.

The licensee stated that, beyond showing that small surface flaws would not develop to through-wall flaw, the FCG evaluation demonstrates that the growth of a flaw will be very slow. The licensee also explained that the analysis further supports the justification that the potential FCG for the SI piping would be insignificant between the time when leakage reaches 5 gpm and the time that the reactor would be shut down due to the leakage, consistent with the acceptability basis of the LBB analysis.

The NRC staff finds that the FCG analysis results for the SI piping with postulated surface flaws adequately support that: (a) the potential FCGs for the 10-year and 60-year periods are insignificant, and (b) the potential FCGs would not affect the crack stability and LBB applicability for the subject SI piping.

In addition, WCAP-18516, Revision 1, addresses a plant-specific FCG analysis for the SI piping with postulated initial through-wall cracks under the design transients. Specifically, the FCG

analysis considered crack growth from the crack size (length) that is associated with 5 gpm leakage, through the crack size that is associated with 10 gpm leakage, to the critical crack size in the SI piping system. The FCG analysis results concluded the FCG from the crack size of 10 gpm leakage to the critical crack size would take approximately 4 years. In its review, the NRC staff finds that there would be a sufficient time to safely shut down the reactor before the leakage crack grows to the critical crack size due to the fatigue mechanism. The NRC staff also finds that the FCG analysis confirms that the FCG is not a concern for the crack stability and LBB applicability of the subject SI piping because the fatigue crack growth from the leakage crack would take approximately 4 years to reach the critical crack and there would be a sufficient time to safely shut down the reactor prior to piping rupture.

WCAP-18516, Revision 1, also provides the FCG analysis that is bounding for the accumulator, RHR, and pressurizer surge lines with postulated circumferential inner-surface cracks. The WCAP report indicates that the FCG analysis uses representative piping geometry, loading conditions, and transient cycle numbers for the piping lines. In particular, the FCG analysis for the pressurizer surge lines includes representative thermal stratification transients for the piping line. The FCG analysis results confirmed that the postulated initial surface cracks would not grow to through-wall cracks and that the FCG is insignificant for 60 years of operation.

The NRC staff finds that the FCG analyses with postulated surface cracks in the accumulator, RHR, and pressurizer surge lines support that the postulated surface cracks and their FCG would not result in through-wall cracking during the plant operation and that the potential FCG would not affect the crack stability and LBB applicability of the subject piping because the FCG is insignificant for 60 years of operation and would not cause through-wall cracks.

With respect to the potential for high cycle fatigue, the licensee stated that pump vibrations during operation would result in high cycle fatigue loads in the piping system. During operation, an alarm signals the exceedance of the reactor coolant pump shaft vibration limits. Additionally, the licensee stated that field vibration measurements have been made on the reactor coolant loop piping in a number of plants during hot functional testing. The stresses in the elbow below the reactor coolant pump have been found analytically to be very small – between 2 and 3 ksi (thousands pounds per square inch) at the highest. Field measurements on typical PWR plants indicate vibration stress amplitudes less than 1 ksi. When translated to the branch lines connected to the RCS primary loops, these stresses would be even lower, well below the fatigue endurance limit for the materials of the subject piping, and would not result in fatigue crack growth.

The NRC staff finds that low cycle and high cycle fatigue is not a potential source of pipe rupture for the subject piping because the licensee clarified that: (a) the cycles of the low cycle fatigue meet the allowable cycle limit in accordance with the B31.1 code, (b) the low cycle fatigue in the RCS is also managed by the fatigue TLAA and Fatigue Monitoring Program, (c) the FCG analyses for the subject piping support that the potential FCGs are insignificant and do not affect crack stability and LBB applicability, and (d) the high cycle fatigue due to pump vibration is insignificant.

3.2.2.2 Thermal Stratification

The licensee indicated that thermal stratification occurs when hot and cold layers of water exist simultaneously in a section of piping and can cause significant thermal loads. Therefore, fatigue cracking can occur in piping lines susceptible to thermal stratification and associated stress cycles. The licensee also indicated that, based on industry experience with thermal stratification

or temperature oscillations in systems attached to the RCS, the NRC issued Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," dated June 22, 1988 (ADAMS Accession No. ML031220144). The licensee provided its evaluation regarding the susceptibility of the accumulator, SI, and RHR lines to thermal stratification as summarized below.

The licensee stated that Bulletin 88-08 requested licensees to review systems connected to the RCS to determine whether unisolable sections of piping connected to the RCS can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves and that were not evaluated in the design analysis of the piping. The licensee confirmed that it identified three potential locations of concern at each Salem unit. The licensee further stated that these locations are not within the scope of the LAR and that the concern of each location was addressed in response to Bulletin 88-08, and its supplements, in the current licensing bases (e.g., by valve leak rate testing and design changes).

The NRC staff notes that the licensee previously addressed the thermal stratification concerns of the subject branch lines connected to the RCS in response to Bulletin 88-08. The NRC staff finds that those susceptible locations are not in the scope of the LAR. Therefore, fatigue cracking due to thermal stratification is not an active degradation mechanism for the subject accumulator, SI, and RHR piping lines.

In its evaluation for the pressurizer surge line, the licensee explained that the piping line is subject to thermal stratification due to water temperature differences in the line during performance of its design functions, as addressed in the NRC issued Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," dated December 20, 1988 (ADAMS Accession No. ML031220290). The licensee also stated that, as part of the license renewal for 60 years of operation, a plant-specific structural evaluation of the surge lines was performed to address the thermal stratification concern. The licensee further stated that the evaluation is described in WCAP-12914, "Structural Evaluation of Salem Nuclear Plant Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," Revision 1. In addition, the licensee confirmed that the evaluation demonstrates that the pressurizer surge lines meet the criteria for stress limits and fatigue usage in ASME Code, Section III, "Rules for Construction of Nuclear Facility Components." The licensee also clarified that the stress and fatigue evaluation of the pressurizer surge line was performed as part of the June 2011 license renewal of Salem, Unit Nos. 1 and 2, which the NRC staff approved (ADAMS Accession No. ML11166A135).

The NRC staff finds that fatigue cracking due to thermal stratification is sufficiently addressed for the subject pressurizer surge line because the licensee confirmed that the ASME Code, Section III fatigue usage criteria are met, as demonstrated for the license renewal of Salem, Unit Nos. 1 and 2.

3.2.3 Brittle Fracture and Cleavage-Type Failure

The licensee stated that brittle fracture for stainless steel material occurs when the operating temperature is approximately minus 200 °F. The operating temperatures of the accumulator, RHR, SI, and pressurizer surge lines are higher than 120 °F, and therefore, brittle fracture is not a concern for potential failure of these lines. The licensee further stated that brittle cleavage-type failures are not a concern based on the operating temperatures and the stainless steel material used in the subject piping lines.

The NRC staff finds that brittle fracture or cleavage-type failure is not an active degradation mechanism for the subject piping because the operating temperatures of the subject piping are not within the temperature range that would cause brittle fracture.

3.2.4 Creep

The licensee stated that the maximum operating temperature of the piping lines in the LAR scope is approximately 650 °F (i.e., pressurizer surge line temperature), which is below the temperature at which creep damage would occur in stainless steel piping. The NRC staff recognizes that the operating temperature of the subject piping is well below the temperature (800 °F) that would cause creep damage in stainless steel material. Therefore, creep damage is not an active degradation mechanism.

3.2.5 Wall Thinning

The licensee stated that wall thinning by erosion and erosion-corrosion should not occur in the accumulator, RHR, SI, and pressurizer surge line piping because of the low velocity of the flow, typically less than 1.0 foot per second (ft/sec), and the stainless steel material used for the piping, which is highly resistant to these degradation mechanisms. The licensee identifies that as discussed in NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," a study on pipe cracking in PWR piping reported only two incidents of wall thinning in stainless steel pipe, and these were not in the accumulator, RHR, SI, or pressurizer surge lines. The licensee also indicated that these evaluation results are consistent with its procedure for the Flow Accelerated Corrosion Program, which screens out stainless steel piping based on the material's resistance to erosion and erosion-corrosion mechanisms.

The NRC staff finds that wall thinning is not considered as an active degradation mechanism for the subject piping because of the low velocity flow in the subject piping and the piping fabrication material is resistant to wall thinning. The NRC staff also notes that operating experience has not shown wall thinning in the subject piping lines.

3.2.6 Water Hammer

The licensee indicated that the potential for water hammer in the subject piping lines is low because they are designed and operated to preclude voiding conditions within the normally coolant-filled lines. The licensee stated that to ensure dynamic system stability, reactor coolant conditions are stringently controlled. The coolant temperature during normal operation is maintained within a narrow range by the control rod positions. The pressure is also controlled within a narrow range for steady-state conditions by the pressurizer heaters and pressurizer spray. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters (namely system resistance and the reactor coolant pump characteristics) are controlled in the design process.

Additionally, the licensee explained that Westinghouse performed instrumentation and monitoring activities to verify the flow and vibration characteristics of the RCS and the connected auxiliary lines. The licensee's preoperational testing and operating experience have verified that the Westinghouse approach is effective.

The NRC staff notes that the subject lines are designed and operated to minimize a water hammer event. Therefore, the NRC staff finds that a significant water hammer event would not likely occur in the water solid pipes during normal operation.

3.2.7 Conclusion on the Screening Based on Applicable Degradation Mechanisms

Based on the above screening evaluation, the NRC staff finds that accumulator, RHR, SI, and pressurizer surge lines in the scope of the LAR are not subject to any active degradation as specified in SRP 3.6.3. In addition, the licensee confirmed that the ISI examination records for the subject piping show no relevant indications in the subject piping. Therefore, the NRC staff concludes that, given the absence of any active degradation mechanisms and recordable indications in the subject piping, the licensee's LBB analysis meets the acceptance requirements for a fracture mechanics analysis to be used for the determination of the probability of pipe failure, consistent with GDC 4 (discussed above in Section 2.3).

3.3 Limit Load Analyses

3.3.1 Material Properties

SRP Section 3.6.3, subsection III.11, specifies that the LBB analysis should identify the types of materials and materials specifications used for, in part, base metal and weldments. This subsection also specifies that the licensee should provide the material properties, including toughness, tensile data, and long-term effects such as thermal aging.

As previously discussed, the licensee reported that the subject piping is fabricated with stainless steel materials. Specifically, the materials for fabrication of the subject piping are A376 TP316 stainless steel (seamless pipes), A403 WP316 stainless steel (wrought fittings), and A182 F316 stainless steel (branch nozzles at the reactor coolant loops). The licensee also confirmed that the subject piping lines are fabricated with forged product forms that are not susceptible to fracture toughness degradation due to thermal aging. The weld processes used in the subject piping are SAW and SMAW processes.

The licensee indicated that the tensile properties of the piping materials in the Certified Material Test Reports were used to estimate the material properties at operating temperatures for the subject piping lines. In the estimations, the licensee used the ratios of the tensile properties between different temperatures, which were based on the data in the 2007 Edition with 2008 Addenda of ASME Code, Section II, "Materials." The licensee estimated the tensile properties of piping materials at operating temperatures by using the code property ratios and performing linear interpolations with Certified Material Test Reports tensile properties to consider the temperature effects on material properties.

The licensee also interpolated material modulus of elasticity from ASME Code values to estimate the property values at the operating temperatures considered. The Poisson's ratio was taken as 0.3. The yield strengths, ultimate strengths, and elastic moduli for the materials of each subject piping are provided in Section 4.0 of the WCAP-18248, WCAP-18249, WCAP-18253, and WCAP-18261 reports.

Based on engineering judgment, the NRC staff finds the licensee's approach acceptable because: (a) the licensee accounted for the temperature effects on material properties by interpolating material property data between different temperatures; and (b) the licensee

adequately used the tensile property ratios based on code property data to estimate the operating temperature properties of the subject piping.

3.3.2 Load Combinations

SRP Section 3.6.3, subsection III.1, specifies that the LBB analysis should use design basis loads that are based on the as-built piping configuration, as opposed to the design configuration. As described in the WCAP reports for the subject piping lines, the licensee stated that the LBB analysis used the as-built piping configurations and the associated piping loads.

SRP Section 3.6.3, subsection III.11.C.v, addresses the level of conservatism that needs to be applied to the load calculations in the crack stability analysis. The SRP indicates that, if the deadweight, thermal expansion, pressure, safe shutdown earthquake, and seismic anchor motion loads are combined based on the individual absolute values of the loads (i.e., absolute sum load combination method), no additional margin may be applied to the limiting load calculation.

In the crack stability analysis, the licensee used the absolute sum load combination method in accordance with the guidance in the SRP. The licensee also considered the bending and torsional moments to obtain the limiting total applied moment. In addition, the licensee calculated the applied moment based on the square root of the sum of squares of the bending and torsional moments, which is consistent with SRP Section 3.6.3, subsection III.11.C.v.

With respect to the loading evaluation for the pressurizer surge lines, the licensee stated that, because thermal stratification can cause large stresses during heatup and cooldown, a review of the stratification stresses was performed to identify the upper bound loadings. Accordingly, the licensee determined the thermal stratification loadings for normal operation and faulted loading conditions. The licensee also considered the probable combinations of the normal operation conditions and faulted conditions, including different scenarios of thermal stratification. In addition, the combinations of the normal and faulted loading conditions (also called loading case combinations) were used in the estimation of leakage crack sizes and critical crack sizes. The NRC staff finds that these loading case combinations adequately account for thermal stratification loads and represent the normal and faulted locating conditions because the licensee considered the bounding loads of the relevant transients that can cause thermal stratifications and included both normal and faulted conditions in the load combinations.

The NRC staff finds that the licensee's load combinations in the crack stability analysis of the subject piping are acceptable because: (a) the licensee used the absolute sum load combination method; (b) the licensee appropriately considered the maximum loads under the faulted conditions in the limiting load combinations (including deadweight, thermal expansion, safe shutdown earthquake, and seismic anchor motion loads); (c) the calculations of the total moment considered the bending and torsional moments; (d) thermal stratification stresses are considered in the loading case combinations for the normal operation and faulted conditions of the pressurizer surge lines; and (e) these methods are consistent with the guidance in SRP Section 3.6.3, subsections III.1 and III.11.C.v.

3.3.3 Leakage Crack Size Calculation

SRP Section 3.6.3, subsection III.11.C.iii, specifies that the estimated leak rate from the leakage crack during normal operation should be 10 times greater than the minimum leak rate that the RCS leakage detection systems can detect.

The licensee stated that the RCS pressure boundary leak detection system can detect a leak rate of 0.5 gpm. The licensee also stated that, for the 6-inch SI lines evaluated in WCAP-18248, a leakage of 5 gpm and a detection capability of 0.5 gpm are used in order to meet the margin in SRP 3.6.3, subsection III.11.C. The licensee further indicated that the accumulator, RHR, and surge lines are larger diameter piping with higher leakage rates, and therefore, the LBB analysis for the piping lines uses a leakage rate of 10 gpm with a detection capability of 1 gpm. The NRC staff's evaluation regarding the leakage detection capability is documented in Section 3.4 of this safety evaluation.

The licensee stated that, for the single-phase flow cases with lower coolant temperature, leakage rate is calculated by considering the frictional pressure losses, including the flow passage, inlet, and outlet losses. The licensee explained that the flow of hot pressurized coolant through an opening to the outside of the subject piping at atmosphere pressure may involve flashing that can result in a two-phase choked flow. Using an assumed leakage flow rate, the licensee calculated the frictional pressure drop based on the friction factor of the leakage path considering the relative roughness of the crack surface. The licensee also calculated the two-phase flow pressure drop to estimate the total pressure drop through the leakage path. The assumed flow rate was adjusted iteratively until these calculations resulted in the total pressure drop value for the leakage flow.

In addition, the licensee stated that it used the crack opening area obtained by the method from the following reference: Section II-1, "The Effects of Shell Corrections on Stress Intensity Factors and the Crack Opening Area of Circumferential and a Longitudinal Through-Crack in a Pipe," of NUREG/CR-3464, "The Application of Fracture-Proof Design Methods Using Tearing-Instability Theory to Nuclear Piping Postulating Circumferential Through-Wall Cracks," dated September 1983.

In its review, the NRC staff finds that the estimated size (length) of the leakage crack is large enough so that leakage from the flaw during normal operation would be 10 times greater than the minimum leakage that the RCS detection system is capable of detecting. Therefore, the NRC staff finds that the licensee's approach is consistent with the guidance in SRP Section 3.6.3, subsection III.11.C.iii. The NRC staff also noted that the licensee's methods used to estimate the leakage rates and leakage crack sizes for given leakage detection limits are consistent with those used in the existing LBB analysis for the primary coolant loops, which constitute the current licensing basis of Salem, Unit Nos. 1 and 2 (Reference: WCAP-13660, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Salem Generating Station Units 1 and 2," dated May 1993 (ADAMS Package Accession No. ML18100A462).

3.3.4 Crack Stability Analysis

SRP Section 3.6.3, subsection III.11.C, describes how the critical crack sizes should be calculated. SRP Section 3.6.3, subsection III.11.C.iv, specifies that a crack stability analysis should be performed to demonstrate that the leakage crack size will not become unstable by comparing the leakage crack size to the critical crack size. Specifically, a margin of 2 should exist between the leakage crack size and critical crack size.

The licensee derived the critical crack sizes at the critical locations in accordance with the guidance in SRP Section 3.6.3, as discussed below. The licensee calculated the highest faulted stresses and identified the corresponding weld location node for each welding process type in

the subject piping to determine the critical locations. Therefore, the critical locations are established based on the bounding loads and piping material properties.

In the crack stability analysis, the licensee also used the limit load method to predict the critical crack size for the critical locations in the subject piping. The failure criterion has been obtained by requiring equilibrium of the section containing a through-wall circumferential crack. The applied loads are calculated in consideration of internal pressure, axial force, and imposed moments. The limiting moment for the analyzed pipe is calculated based on the flow stress, axial force, pipe dimensions, and crack size and configuration. For the limit load method, the licensee also multiplied the pipe loads by the Z factor because the shop and field welds were fabricated with SMAW and SAW processes. The licensee stated that it derived Z factors for SMAW and SAW in accordance with SRP 3.6.3.

The NRC staff noted that the crack stability analysis using the limit load method confirmed the stability of the critical cracks in the critical locations. In addition, the licensee's analysis confirms that the critical crack size is at least twice the leakage crack size at the critical locations, thereby demonstrating a margin of 2 between the leakage crack size and critical crack size.

The NRC staff finds that the crack stability analysis is acceptable because: (1) the limit load method was used with relevant Z factors for the SMAW and SAW locations, (2) a safety margin of 2 is demonstrated between the leakage crack size and the critical crack size, and (3) the licensee's methodology is consistent with the guidance in SRP Section 3.6.3. Therefore, the NRC staff finds that the licensee has demonstrated crack stability in the subject piping by performing the limit load analysis appropriately.

3.3.5 Power Uprate

In a previous licensing action, the licensee stated that the 1.4 percent measurement uncertainty recapture (MUR) power uprate resulted in an increase in hot-leg temperature of 0.5 °F and a decrease in cold leg temperature of 0.5 °F (ADAMS Accession No. ML011350051). As part of the subject LAR, the licensee explained that the impact of these changes on RCS piping loads was negligible and that the uprate did not result in changes to nuclear steam supply system design transients. The licensee also indicated that the operating temperatures used in the LBB analysis of the subject piping are consistent with the power uprate parameters. The NRC staff finds the licensee's evaluation regarding the MUR condition is acceptable because: (1) the LBB analysis used the updated operating temperatures of the power uprate, and (2) the MUR power uprate caused negligible effects on piping loads in the scope of LBB analysis.

3.3.6 License Renewal

In relation to the FCG analysis discussed above, the licensee indicated that the LBB analysis for the subject piping considers time dependencies of crack growth since the FCG analysis involves a time-limited assumption. The licensee stated that the FCG analysis for the subject piping bounds the transient projections for the 60-year operation. The NRC staff finds the licensee's evaluation acceptable because the licensee has adequately performed the FCG analysis and confirmed the validity of the LBB analysis for the renewed license term (i.e., 60 years of operation).

3.3.7 NRC Staff's Confirmatory Analysis

The NRC staff performed a confirmatory analysis to check the adequacy of the licensee's analysis results. The confirmatory analysis used the PICEP computer code (Reference: Electric Power Research Institute (EPRI) NP-3596-SR, Revision 1, "PICEP: Pipe Crack Evaluation Program," dated December 1987). The analysis evaluated the following critical locations: (1) FW-2-RH-182-2 (accumulator line), (2) FW-1-RH-134-116A (RHR line), (3) 2-RH-113-A (SI line), and (4) FW-1-RC2-19A (pressurizer surge line).

The PICEP code conducts a limit load analysis with Z factors applied to the loads, which is consistent with the licensee's analysis. In the analysis, the NRC staff finds that the critical crack sizes determined by the licensee are in good agreement with those calculated by the PICEP code. In addition to the PICEP code analysis, the NRC staff conducted independent calculations to estimate the critical crack sizes based on a limit load analysis without using the PICEP code. The additional independent calculation results agreed with the licensee's estimations and PICEP code analysis results for the critical crack sizes.

The NRC staff also used the PICEP code to estimate the leakage crack sizes and check the margin between the leakage crack size and the critical crack size. The confirmatory analysis used the same leakage rates with those used in the licensee's analysis (i.e., 5 gpm for the SI piping and 10 gpm for the other piping). In the confirmatory analysis, the NRC staff finds that a margin of 2 exists between the leakage crack size and the critical crack size.

3.4 RCS Leakage Detection System Capability

This section addresses the NRC staff's evaluation of the LAR regarding RCS leakage detection methods, capabilities, administrative controls, and TSs. Section 3.4, "RCS Leakage Detection Capability," of Enclosure 1 to the LAR describes RCS leakage detection for the LBB applications. Section 3.4 of Enclosure 1 to the LAR states that:

> Recognizing that no single method of leakage detection is capable of rapidly detecting every potential RCS leak, PSEG uses diverse means of measuring and evaluating RCS leakage at Salem. These methods include TS-required leakage detection measurement systems and RCS inventory balance surveillance requirements, leak detection methods that are not required by TS, and administrative controls to monitor and trend RCS leakage to ensure that low levels of RCS leakage are detected and appropriate corrective actions are initiated.

Additionally, in Section 3.4.1 of Enclosure 1 to the LAR, the licensee describes that the following RCS leakage detection systems are required by the current plant TSs:

- containment atmosphere particulate radioactivity monitoring system,
- containment sump level monitoring system, and
- either containment fan cooler condensate flow rate or containment gaseous radioactivity monitoring system

The licensee also provided more detailed information for the above detection methods being used in the TSs in Section 3.4.2, "Containment Atmosphere Particulate Monitor (1R11A, 2R11A)"; Section 3.4.3, "Containment Pocket Sump Level Monitoring System"; Section 3.4.4, "Containment Fan Cooler Condensate Flow Rate"; and Section 3.4.5, "Containment

Atmosphere Noble Gas Monitor (1R12A, 2R12A)," in Enclosure 1 of its LAR. The licensee also described additional non-TS required methods for diverse leak detection capabilities and administrative controls in Section 3.4.6, "Water Inventory Balance," and Section 3.4.7, "Other RCS Leakage Detection Methods and Administrative Controls".

The NRC staff compared the above RCS leakage detection design with the guidance in RG 1.45, Revision 1, and finds that the design is consistent with RG 1.45 as the licensee identifies diverse detection methods.

The SRP, Section 3.6.3, Revision 1, states that plant-specific leakage detection systems inside containment should be equivalent to those in RG 1.45, Revision 1, which states that plants should use leakage detection systems with a response time of no greater than 1 hour for a leakage rate of 1 gpm. The applicability of the current TS limit of 1 gpm for unidentified leakage is discussed further below.

Section 3.4 of Enclosure 1 to the LAR indicates that, as determined by the fracture mechanics evaluations for the Salem LBB application, the limiting leakage flaw is in the 6-inch SI piping and results in a leak rate of 5 gpm, corresponding to a detection capability of 0.5 gpm based on the margin of 10 in SRP Section 3.6.3. RCS leakage detection capability requirements for this LAR are based on the leakage flaw size that produces the lowest leak rate. Unit No. 1 TS 3/4.4.6.2 and Unit No. 2 TS 3/4.4.7.2 each include limitations on RCS leakage during plant operation, including zero allowable pressure boundary leakage and a 1 gpm limit for unidentified leakage.

Section 4.1, "Applicable Regulatory Requirements/Criteria," of Enclosure 1 to the LAR states that the licensee is not proposing any reduction in the TS unidentified leakage limit because the 1 gpm limit remains adequate to assure timely identification of RCPB degradation, measurement capability is sufficient to ensure that RCS leakage can be detected well in advance of a through-wall flaw propagating to a pipe rupture, and the capabilities supporting the LBB analysis assumptions may be appropriately controlled as UFSAR information. The licensee further states that the adequacy of the current TSs is supported by the margins used in the LBB evaluations (i.e., (1) margin factor of 10 between leakage crack flow rate and leakage detection capability, and (2) a factor of 2 between leakage crack size and critical crack size). These margins offset uncertainties associated with leakage detection and prediction. The 1 gpm limit for unidentified leakage is also shown to be conservative for the LBB evaluations by the fatigue crack growth evaluations in WCAP-18516. Evaluation of the 6-inch SI lines shows that it would take nearly 4 years of leakage in a 6-inch SI line to grow to the critical crack size. There are no proposed changes to the Salem TSs associated with this LBB application change.

Section 3.4.7 of Enclosure 1 to the LAR states that to support early detection of increases in unidentified leakage, the NRC and industry developed criteria for monitoring and trending leakage at low levels and establishing action levels for increased leakage. The licensee stated that it has incorporated guidance from NRC Inspection Manual Chapter 2515, "Light-Water Reactor Inspection Program – Operations Phase," Appendix D (ADAMS Accession No. ML092220484), and WCAP-16465-NP into Salem procedures in order to provide early detection and minimize potential consequences associated with RCS leakage. The licensee monitors and trends operational RCS leakage, including unidentified leakage, and has three tiers of procedures starting from unidentified RCS leakage of 0.1 gpm, 0.15 gpm, and 0.3 gpm, respectively. The leakage detection system can detect a leakage of 0.5 gpm.

The NRC staff finds that among the four newly proposed LBB lines (the 6-inch SI lines, 10-inch ECCS accumulator lines, 14-inch RHR lines, and 14-inch pressurizer surge lines), that the

6-inch SI line is the most limiting case for the determination of TS unidentified leakage limit. This is because of its smaller size, which corresponds to smallest critical crack size and associated TS limit.

The NRC staff finds the current TS limit of 1 gpm for unidentified leakage continues to be acceptable based on the following: (1) a margin of 10 exists between the leak rate from the leakage flaw size and the RCS leakage detection system capability; (2) a margin of 2 exists between the critical flaw size and the leakage flaw size; (3) the leakage detection system can detect a leakage of 0.5 gpm; (4) tiered procedures exist for monitoring, trending, and taking appropriate actions to manage the leakage starting from 0.1 gpm unidentified RCS leakage; (5) the leakage surveillance requirements specified in Unit No.1 TS 4.4.6.2 and Unit No. 2 TS 4.4.7.2 are sufficient to ensure the adequate leakage detection from the reactor coolant system; and (6) the TS limit of 1 gpm for unidentified leakage specified in Unit No. 1 TS 3.4.6.2 and Unit No. 2 TS 3.4.7.2 ensures that the reactor can be shut down safely prior to pipe rupture or any significant impact on the structural integrity of the subject piping. These safety margins are consistent with SRP Section 3.6.3 and the previous safety evaluations for Waterford Steam Electric Station and D. Donald C. Cook Nuclear Plant, as referenced in the LAR. Additionally, the potential growth of the postulated leakage crack will take a very long time (i.e., on the order of years) and, as a result, allow for sufficient time for operator actions.

Based on above, the NRC staff finds the licensee's justification acceptable for maintaining the 1 gpm of the current TS unidentified operation leakage limit. The TS leakage limit of 1 gpm is also consistent with RG 1.45, Revision 1.

Per 10 CFR 50.36(b), the technical specifications will be derived from the analyses and evaluations in the licensee's safety analysis report, although the Commission may include such additional technical specifications as the Commission finds appropriate. Although the licensee requested changes to its safety analysis report, the licensee asserted in its LAR that no conforming changes in the TS were needed. As a threshold matter, the TSs already provide limiting conditions for operation (LCOs) and actions to be taken when the LCOs are not met, as required by 10 CFR 50.36(c)(2)(i). Also, the TS already provide surveillance requirements to assure that the LCOs are met as required by 10 CFR 50.36(c)(3). The licensee considered if the change to its UFSAR would cause one of the criteria under 10 CFR 50.36(c)(2)(ii)(A) and (B) for proposing a new LCO to be met. The licensee concluded that amending the UFSAR would not require a new LCO under 10 CFR 50.36(c)(2)(ii)(A)-(B). The staff agree; the existing LCOs in the TS would continue to state the "lowest functional capability or performance levels of equipment required for safe operation of the facility" as required by 10 CFR 50.36(c)(2)(i). Therefore, no change is needed. Further, the change to the UFSAR did not cause a criterion under 10 CFR 50.36(c)(2)(ii)(A)-(B) to be newly met, thus the licensee did not need to propose a new LCO; the existing LCOs are sufficient. Last, no changes to SRs are needed because the existing SRs will be adequate to assure that the LCOs are met after the UFSAR is updated.

In addition to the acceptability of the current TS leakage limit, the NRC staff finds that the detection of the leakage from newly proposed LBB lines has no impact on the current TS detection relating to RCS leakage detection systems, action requirements, and surveillance requirements because the leakage detection system and the associated actions can adequately detect and manage a leakage starting from 0.1 gpm. Additionally, compliance with 10 CFR 50.36(c)(2)(i) remains as the TS LCOs are not affected or changed by this LAR because the leakage is stable and bounded by the TS limit, which ensures the leakage would not lead to critical cracks.

3.5 Technical Conclusion

On the basis of its review of the LAR, as supplemented, the NRC staff finds that, for the subject accumulator, RHR, SI, and pressurizer surge line piping, the licensee has demonstrated the following: (1) the screening criteria of SRP Section 3.6.3 are satisfied in the evaluation of applicable degradation mechanisms, (2) a margin of 10 exists between the calculated leak rate from the postulated leakage crack sizes and the RCS leakage detection system capability; (3) a margin of 2 exists between the leakage crack sizes and the critical crack sizes; (4) the critical cracks were calculated conservatively in consideration of the bounding material properties and load conditions in the limit load analyses; (5) the licensee's LBB methods are consistent with the guidance in SRP Section 3.6.3; (6) the leakage cracks are stable so that the structural integrity of the piping is maintained; (7) the potential FCG in the subject piping does not affect the validity of the LBB analysis; and (8) the potential FCG from the postulated leakage crack to the critical crack involves a very long time period such that the leakage from a through-wall crack would be detected before piping rupture.

Additionally, the NRC staff finds that the licensee's analysis has demonstrated that the subject accumulator, RHR, SI, and pressurizer surge line piping has an extremely low probability of rupture. As described in Section 3.1 of this SE, the licensee has limited the scope of the LBB application to certain RCS piping segments associated with the accumulator, RHR, SI, and pressurizer lines.

Based on the evaluation above, the NRC staff has concluded that the Salem LBB leakage detection is consistent with (1) the requirement of 10 CFR 50.36(c)(2)(i), and (2) the guidance of RG 1.45, Revision 1, for RCS leakage detection capabilities. Furthermore, the NRC staff concludes that the justifications for using the current TS leakage limit for the newly proposed LBB applications of the four RCS lines are acceptable because the leakage detection system can detect a leakage of 0.5 gpm and tiered procedures exist for monitoring, trending, and taking appropriate actions to manage the leakage starting from 0.1 gpm unidentified RCS leakage.

Pursuant to GDC 4 of Appendix A to 10 CFR Part 50, the NRC staff concludes that the licensee is permitted to exclude consideration of the dynamic effects associated with the postulated rupture of the subject accumulator, RHR, SI, and pressurizer surge line piping from the current licensing basis at Salem, Units Nos. 1 and 2.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments on October 13, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 amendments involve 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 Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, published in the *Federal Register* on July 7, 2020 (85 FR 40692), and there has been no public comment on such finding. Accordingly, the

amendments meet 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 amendments.

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.

Principal Contributors: S. Min
C. Li

Date: February 23, 2021

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 336 AND 317 RE: LEAK-BEFORE-BREAK FOR ACCUMULATOR, RESIDUAL HEAT REMOVAL, SAFETY INJECTION, AND PRESSURIZER SURGE LINES (EPID L-2020-LLA-0088) DATED FEBRUARY 23, 2021

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