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

August 20, 2021

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 304 AND 304 RE: LEAK-BEFORE-BREAK FOR PRESSURIZER SURGE, RESIDUAL HEAT REMOVAL, SAFETY INJECTION ACCUMULATOR, REACTOR COOLANT SYSTEM BYPASS AND SAFETY INJECTION LINES (EPID L-2020-LLA-0255)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 304 to Renewed Facility Operating License No. DPR-32 and Amendment No. 304 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station (Surry), Unit Nos. 1 and 2, respectively. These amendments consist of changes to the Surry, Unit Nos. 1 and 2, Updated Final Safety Analysis Report in response to your application dated October 22, 2020, as supplemented by letter dated March 24, 2021.

The amendments made changes to the Surry Updated Final Safety Analysis Report to apply the leak-before-break methodology to specific portions of accumulator piping, residual heat removal piping, safety injection piping, pressurizer surge line piping, and reactor coolant system loop bypass piping, to eliminate the dynamic effects of postulated pipe ruptures in specific portions of the auxiliary system attached to the primary reactor coolant system.

A copy of the related safety evaluation is also enclosed. The Commission's monthly *Federal Register* notice will include the notice of issuance.

Sincerely,

/RA/

Vaughn V. Thomas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 304 to DPR-32
2. Amendment No. 304 to DPR-37
3. Safety Evaluation

cc: Listserv

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 304
Renewed License No. DPR-32

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated October 22, 2020, as supplemented by letter dated March 24, 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, by Amendment No. 304, Renewed Facility Operating License No. DPR-32, is hereby amended to authorize the change to the Updated Final Safety Analysis Report (UFSAR) as requested by letter dated October 22, 2020, as supplemented by letter dated March 24, 2021, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

3. The 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

Glenn E. Miller

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Date: 2021.08.20 14:08:24 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: August 20, 2021

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 304 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Deleted by Amendment 65

F. Deleted by Amendment 71

G. Deleted by Amendment 227

H. Deleted by Amendment 227

I. Fire Protection

The licensee shall implement and maintain in effect the provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER dated September 19, 1979, (and Supplements dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and

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WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 304
Renewed License No. DPR-37

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated October 22, 2020, as supplemented by letter dated March 24, 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, by Amendment No. 304, Renewed Facility Operating License No. DPR-37, is hereby amended to authorize the change to the Updated Final Safety Analysis Report (UFSAR) as requested by letter dated October 22, 2020, as supplemented by letter dated March 24, 2021, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

3. The 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

Glenn E. Miller

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Date: 2021.08.20 14:09:06 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: August 20, 2021

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such by product and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power Levels not in excess of 2587 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 304 are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the Requirements of the Technical Specifications.

E. Deleted by Amendment 54

F. Deleted by Amendment 59 and Amendment 65

G. Deleted by Amendment 227

H. Deleted by Amendment 227

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSE AMENDMENT REQUEST REGARDING APPLICATION OF

LEAK-BEFORE-BREAK METHODOLOGY

FOR REACTOR COOLANT SYSTEM BRANCH PIPING

AMENDMENT NO. 304 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 304 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNITS 1 AND 2,

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated October 22, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20296A623), as supplemented by letter dated March 24, 2021 (ADAMS Accession No. ML21084A273), Virginia Electric and Power Company (Dominion Energy Virginia, licensee) submitted a license amendment request (LAR) for Surry Power Station (Surry), Units 1 and 2. The proposed amendment would apply the leak-before-break (LBB) methodology to specific portions of accumulator piping, residual heat removal (RHR) piping, safety injection (SI) piping, pressurizer surge line piping, and reactor coolant system (RCS) loop bypass piping. The subject piping within the scope of the LAR is the branch (auxiliary) piping attached to the primary reactor coolant loop (RCL) piping.

The LAR proposes to eliminate the dynamic effects of postulated pipe ruptures in the subject piping from the design basis of Surry Units 1 and 2. The licensee submitted the LAR in accordance with General Design Criterion (GDC) 4, “Environmental and dynamic effects design bases,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic licensing of production and utilization facilities.” The request does not propose changes to the technical specifications (TS). Attachment 2 to the licensee’s request contains sensitive unclassified non-safeguards information. As the licensee requested, the U.S. Nuclear Regulatory Commission (NRC) has withheld the sensitive information from public disclosure pursuant to 10 CFR 2.390, “Public inspections, exemptions, requests for withholding.”

The supplemental letter provided additional information, did not expand the scope of the application that was originally proposed, and did not change the NRC staff's proposed determination of no significant hazards consideration as published in Volume 86 of the *Federal Register*, page 7118 (86 FR 7118), on January 26, 2021.

2.0 REGULATORY EVALUATION

2.1 Proposed Change and Related Systems

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 catastrophic pipe rupture and to ensure that the probability of pipe rupture is extremely low. In the LBB concept, the RCS leakage detection system will detect small through-wall cracks and the associated leakage promptly so that operators can shut down the reactor and take corrective actions before pipe rupture.

Surry Units 1 and 2 are three-loop Westinghouse pressurized-water reactors (PWRs). As described in the Surry Updated Final Safety Analysis Report (UFSAR), Section 4.2.4, "Protection Against Proliferation of Dynamic Effects"; Section 15.6.2, "Reactor Coolant System Supports"; Section 15A.3.3, "Reactor Coolant Loops and Supports"; Section 15A.6, "Reactor Coolant Loop Piping Reanalysis Subsequent to Leak Before Break and Snubber Elimination"; and Section 18.3.5.3, "Leak-Before-Break," the existing LBB analyses have demonstrated the probability of rupture of the primary RCL piping is extremely small and, therefore, it is no longer necessary to consider the dynamic effects of such an accident in the RCL piping.

This LAR proposes to expand the application of LBB methodology to include certain portions of the following branch piping attached to the RCL: (1) accumulator lines (also called SI accumulator injection lines) attached to the cold-leg RCL piping, (2) RHR suction line attached to the hot-leg RCL and RHR return line attached to the accumulator injection piping, (3) SI lines attached to the hot-leg and cold-leg RCL piping, (4) pressurizer surge lines attached to the hot-leg RCL piping, and (5) RCS loop bypass lines attached to the hot-leg and cold-leg RCL piping. Specifically, the scope of the LBB methodology would include the entire pressurizer surge lines and entire RCL bypass lines. For the other branch lines (i.e., accumulator, RHR, and SI lines), the scope of the LBB methodology is limited to the piping segments from the RCL to the first pressure isolation valve.

The expanded scope of the LBB methodology described in the LAR would eliminate analysis of the dynamic effects of postulated rupture of these specific portions of RCS branch piping. Approval of this request to expand the scope of LBB would also affect information in Sections 4.2.4, 15.6.2, 15A.3.3, 15A.6, and 18.3.5.3 of the Surry UFSAR. The licensee stated that, following NRC approval, it will revise the Surry UFSAR to reflect the application of LBB methodology to the subject piping lines.

The licensee's LBB analysis relies on the ability to detect unidentified RCS leakage and take appropriate actions before the TS limit to prevent pipe rupture.

2.2 Current Technical Specifications Requirements

The requirements related to the content of the TS are contained in 10 CFR 50.36, "Technical specifications," which requires, in part, that the TS include limiting conditions for operation (LCOs). The following criteria defined by 10 CFR 50.36(c)(2)(ii) are relevant to determining

whether capabilities related to reactor coolant pressure boundary (RCPB) leakage detection should be included in the TS LCOs:

(A) 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.

(B) 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.

The TS for Surry Units 1 and 2 require periodic verification that RCS leakage is within limits and that leakage detection instrumentation is operable. Specifically, existing TS Surveillance Requirement (SR) 4.13 requires operators verify that RCS operational leakage is within the limits specified in TS 3.1.C, "RCS Operational LEAKAGE," by performance of RCS water inventory balance at the frequencies specified in TS 6.4.S, "Surveillance Frequency Control Program (SFCP)." TS 3.1.C.4 states that "[d]etected or suspected leakage from the Reactor Coolant System shall be investigated and evaluated. At least two means shall be available to detect reactor coolant system leakage. One of these means must depend on the detection of radionuclides in the containment."

The licensee has not proposed changes to the TS.

2.3 Regulatory Requirements and Guidance

The NRC issued construction permits for Surry Units 1 and 2 before May 21, 1971; consequently, Surry Units 1 and 2 were not subject to the requirements in 10 CFR Part 50, Appendix A (see SECY-92-223, "Resolution of Deviations Identified during the Systematic Evaluation Program," ADAMS Accession No. ML003763736 dated September 18, 1992). The conclusion of this effort was that Surry Units 1 and 2 meet the intent of the GDC published in 1967 (draft GDC). The NRC staff reviewed the licensee's request against UFSAR Chapter 4, "Reactor Coolant System"; Chapter 15, "Structures and Construction"; and Chapter 18, "Programs and Activities That Manage the Effects of Aging."

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," the licensee requested to amend its license to apply the LBB methodology to the entire pressurizer surge lines and RCL bypass lines and to specific portions of the accumulator, the RHR, and the SI lines as described in Section 2.1 of this safety evaluation. In addition, 10 CFR Part 50, Appendix A, GDC 4 and GDC 30, "Quality of reactor coolant pressure boundary," are directly applicable to the LAR as further discussed below.

The regulations in GDC 4 state, in part, that structures, systems, and components (SSCs) important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including loss-of-coolant accidents. These SSCs 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 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. UFSAR Sections 4.2.4, 15.6.2, 15A.3.3, 15A.6, and 18.3.5.3 describe the current compliance of Surry Units 1 and 2 with GDC 4.

To meet the requirements of 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 [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition", Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Revision 1, issued March 2007 (ADAMS Accession No. ML063600396), by demonstrating that it does not experience active degradation that can be a source of pipe rupture. 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 system 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 regulations in GDC 30 state that components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical and that the means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

As required by 10 CFR 50.36(c)(2)(i), the TS are to include 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 TS until the condition can be met.

The SRP, Section 3.6.3, provides guidance on screening criteria, safety margins, and analytical methods for the piping systems to be qualified for the application of LBB. SRP Section 3.6.3 states the following:

> Leakage detection systems are evaluated to determine whether they are sufficiently reliable, redundant, and sensitive so that a margin on the detection of unidentified leakage exists for through-wall flaws to support the deterministic fracture mechanics evaluation. The specifications for plant-specific leakage detection systems inside the containment should be equivalent to those in [Regulatory Guide] RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." [Revision 0, ADAMS Accession No. ML003740113].

The Technical Basis for the LBB analysis is provided in NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," issued November 1984 (ADAMS Accession No. ML093170485).

3.0 TECHNICAL EVALUATION

The NRC staff followed the guidance in SRP Section 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 fracture mechanics analysis of the subject piping, as discussed in Section 3.3. In

addition, the NRC staff evaluated the capability of the RCS leakage detection system, as documented in Section 3.4.

As part of the submittal, the licensee provided a nonproprietary report describing the LBB analysis, WCAP-18491-NP, "Technical Justification for Eliminating Auxiliary Piping Rupture as the Structural Design Basis for Surry Units 1 and 2, Using Leak-Before-Break Methodology," Revision 0, issued December 2019 (ADAMS Accession No. ML20296A623).

3.1 Scope of the Leak-Before-Break Analysis

As described in Attachment 1 of the LAR, the licensee's LBB analysis is applied to specific portions of the accumulator, RHR, and SI lines that are connected to the RCLs. The LBB analysis for these piping lines is limited to each piping segment from the RCL up to the first RCS pressure isolation valve. In addition, the scope of the LBB analysis includes the entire pressurizer surge lines and the RCL bypass lines. The licensee analyzed these piping segments to credit the proposed LBB methodology at Surry Units 1 and 2. WCAP-18491, Chapter 3, "Pipe Geometry and Loading," contains detailed piping configurations and sizes. The licensee described the subject piping and related systems as follows:

Accumulator Lines

The accumulator lines are also called SI accumulator injection lines. An accumulator filled with borated water and pressurized with nitrogen is connected to each RCS cold leg. When RCS pressure drops below the nitrogen pressure setpoint, the accumulators discharge their borated water into the RCS. This action provides rapid refilling of the lower core plenum in the event of a large break in the RCS. The LAR includes piping segment ACC-I.

Residual Heat Removal Lines

The RHR system is a low-pressure, low-temperature fluid system that is not used during power operation. The system is designed to operate at pressures less than 450 pounds per square inch gauge (psig) and at temperatures less than 350 degrees Fahrenheit (F). The system is operated during plant cooldown after RCS pressure and temperature are within RHR system limitations. The primary purpose of the RHR system is to remove decay heat energy generated in the reactor core during plant cooldown and refueling operations. During the normal operation of the RHR system, the suction flow of the RHR system is from RCL 1 hot leg, and the discharge flow is to RCLs 2 and 3 through the SI accumulator injection lines. The scope of the LAR includes piping segment RHRs-I.

Safety-Injection Lines

The SI system is designed to provide emergency core cooling following a loss-of-coolant accident for any break in the RCS piping, up to and including the equivalent of a double-ended break in the largest RCS pipe. The SI system is also designed to provide core cooling when the rupture of a control rod drive mechanism housing causes rod ejection, main steam line break, or steam generator tube rupture.

The SI system has three high head SI (HHSI) pumps, which serve as charging pumps during normal operation, and two low head SI (LHSI) pumps. During the injection mode, the HHSI pumps deliver borated water from the refueling water storage tank (RWST) into the cold legs of the RCS. The HHSI pumps maintain RCS pressure in the event of a small break in the RCS.

The LHSI pumps also take suction from the RWST and deliver large quantities of borated water to the cold legs of the RCS when RCS pressure drops below the LHSI pump shutoff head. During recirculation mode, recirculation flow is initially provided to the cold legs of the RCS and is then alternated to provide either hot-leg or cold-leg recirculation flow. The scope of the LAR includes piping segments SI-CL-I and SI-HL-I.

Pressurizer Surge Lines

Pressurizer pressure is transmitted to the remainder of the RCS through the surge line that connects the bottom of the pressurizer with the RCS piping near the outlet of the reactor vessel. The pressurizer surge line connects the bottom of the pressurizer to the hot leg of RCL 3. The scope of the LAR includes piping segments PZR-I and PZR-II.

Reactor Coolant Loop Bypass Line

Each RCL is equipped with a bypass-relief line that connects the loop sides of the loop isolation valves. The line is equipped with an isolation valve that is used to secure flow in the bypass line during normal loop operation. When opened, the bypass line allows operation of a reactor coolant pump (RCP) in an isolated loop by routing RCP discharge through the bypass line to the loop side of the RCS hot-leg isolation valve. The scope of the LAR includes piping segments BP-I and BP-II.

The NRC staff finds that the LAR, as supplemented, has clearly identified the specific portions of the accumulator, RHR, SI, pressurizer (PZR) surge, and RCL bypass (BP) piping lines that are subject to the licensee's LBB analysis, and, therefore, the scope of the LBB analysis is identified appropriately.

3.2 Screening Based on Applicable Degradation Mechanisms

The SRP, Section 3.6.3, Subsection III, specifies that active degradation should not be a potential source that can 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).

In the following sections, the NRC staff evaluates the LBB analysis in accordance with the degradation screening criteria of SRP Section 3.6.3, Subsection III.

3.2.1 Stress Corrosion Cracking

As discussed in Section 2.1, "Stress Corrosion Cracking," of WCAP-18491, the following elements or contaminants in a reactor coolant environment are known to increase the susceptibility of austenitic stainless steel to SCC: oxygen, fluorides, chlorides, 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, and the contaminant concentrations are kept below the thresholds known to be conducive to SCC. Plant operating procedures also include the water chemistry control standards. Therefore, the likelihood of SCC is minimized during plant operation.

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, pressurizer surge, and RCL bypass lines of Surry Units 1 and 2.

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.

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 sections below describe the NRC staff's evaluation on these matters.

3.2.2.1 Low-Cycle and High-Cycle Fatigue

As discussed in Section 2.3, "Low Cycle and High Cycle Fatigue," of WCAP-18491, the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, specifies an explicit fatigue analysis that estimates the cumulative usage factors (CUFs) for Class 1 piping and piping components. However, the licensee indicated that the subject piping was originally designed in accordance with the 1967 Edition of United States of America Standard (USAS) B31.1, "Power Piping," code. Therefore, instead of requiring an explicit fatigue analysis, the subject piping complies with the provision in paragraph 102.3.2 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.

WCAP-18491 also explains 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 at Surry, 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 performed a fatigue time-limited aging analysis (TLAA) using CUFs as part of its subsequent license renewal. This fatigue analysis was conducted in accordance with ASME Code, Section III, Class 1, rules for each of the leading fatigue locations. In addition, environmentally assisted fatigue (EAF) correction factors were calculated and applied to the CUF values developed in the Class 1 fatigue analysis to determine the environmental CUF for subsequent license renewal (i.e., 80 years of operation). The EAF analysis also includes the evaluation of the effects of the reactor water environment on leading fatigue 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," issued February 1995 (ADAMS Accession No. ML031480219).

The NRC staff approved the Surry EAF TLAA in the safety evaluation for the subsequent license renewal of Surry Units 1 and 2 (ADAMS Accession No. ML20052F523). As part of the

aging management program related to the fatigue analyses, the licensee uses the fatigue monitoring program to ensure that the number of occurrences of each critical fatigue transient remains within the limits of the fatigue analysis and the fatigue analysis remains valid. The NRC staff also noted that the fatigue TLAA includes the analysis for the pressurizer surge line and accumulator piping that are within the scope of the LBB analysis.

In addition to the low-cycle fatigue discussed above, the licensee explained that the Materials Reliability Program (MRP) Report MRP-146, “Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines,” Revision 2, Electric Power Research Institute, September 2016, identifies piping systems that may be susceptible to thermal cycling effects. The licensee clarified that the potential for thermal cycling effects on the subject piping is also managed in accordance with the licensee’s inspection activities for the piping.

The NRC staff noted in the SLR SER that the design basis of Surry Units 1 and 2 relies on the USAS B31.1 code provisions for fatigue analyses and the 80-year EAF TLAA for subsequent license renewal that projects CUF values for the leading locations of the RCS piping with environmental effects considered. The NRC staff finds that these fatigue analyses support the evaluation results that (1) degradation due to fatigue will not affect the structural integrity of the subject piping within the scope of this LAR, and (2) potential cracking due to fatigue will be adequately managed for 80 years of operation.

In addition, Section 8.0, “Assessment of Fatigue Crack Growth,” of WCAP-18491 addresses the fatigue crack growth (FCG) analyses of postulated circumferential inner-surface cracks for the subject piping. These FCG analyses address crack growth of postulated initial flaws and may be compared with the CUF analyses discussed above in that the CUF analyses focus on the susceptibility to fatigue initiation.

Section 8.0 of WCAP-18491 explains that these FCG analyses are presented as a defense-in-depth justification for the following criteria: (1) resistance to degradation due to fatigue cycles to confirm that small postulated surface flaws do not become through-wall flaws during the entire operating period of the piping systems, and (2) stability of a through-wall flaw to provide assurance that a leakage flaw can be identified and addressed before growing to a critical flaw size, as supported by the resistance to fatigue degradation evaluated in relation to criterion (1) above.

To evaluate the criteria addressed above, Section 8.1, “Surry Plant-Specific Fatigue Crack Grown Assessments,” of WCAP-18491 describes the plant-specific FCG analyses for the RHR suction lines, accumulator lines, and SI lines. These FCG analyses use the plant-specific pipe geometry with a postulated initial surface flaw at a limiting weld location in each piping system evaluated. In these plant-specific analyses and the representative FCG analyses further discussed below, the depths of the postulated initial surface crack range from 10 percent to 35 percent of the approximate pipe wall thickness. This range is based on the acceptance standards in ASME Code, Section XI, for inspections and detectability of potential cracks. The licensee indicated that the plant-specific FCG analyses estimate the fatigue crack growth under the transient cycles that are expected to occur for the 80-year operation of the Surry plants. These analyses used the circumferential crack orientation that is more limiting for piping rupture than the axial crack orientation.

Table 8-2, “Fatigue Crack Growth Results for the Surry Units 1 and 2 RHR Suction Lines”; Table 8-4, “Fatigue Crack Growth Results for the Surry Units 1 and 2 Accumulator Lines”; and

Table 8-6, “Fatigue Crack Growth Results for the Surry Units 1 and 2 SI Lines,” of WCAP-18491 provide the 80-year crack sizes in the plant-specific FCG analyses for the RHR suction, accumulator, and SI piping lines, respectively. These analysis results confirm that the fatigue crack growth is insignificant. The licensee explained that the FCG evaluation demonstrates the potential crack growth is very slow and that small surface cracks would not develop to a through-wall crack. The licensee also explained that the analyses further support the justification that the potential FCG for these piping lines would be insignificant between the time when leakage reaches 10 gallons per minute (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 plant-specific FCG analysis results for the RHR suction, the accumulator, and the SI piping with postulated surface flaws support the conclusion that (1) the potential FCGs for the 80-year period are insignificant, and (2) the potential FCGs would not affect the crack stability and LBB applicability for the piping lines.

In addition, Section 8.2, “Representative Fatigue Crack Grown Assessments,” of WCAP-18491 provides the representative FCG analyses for the pressurizer surge line of Surry Units 1 and 2. These representative generic FCG analyses are based on a PWR piping system design that can represent the Surry plants. The licensee explained that the generic FCG analysis parameters are bounding for the plant-specific analysis parameters of the Surry plants (e.g., piping geometry, material properties, operating temperature and pressure, piping loads at the evaluated locations, and operating transients and their cycle numbers for the plant life).

As presented in Table 8-8, “Fatigue Crack Growth Results for the Surry Units 1 and 2 Surge Lines”; and Table 8-10, “Fatigue Crack Growth Results for the Surry Units 1 and 2 RHR Return Lines,” of WCAP-18491, the representative FCG analysis results demonstrate that the postulated initial surface cracks in the pressurizer surge line would not grow to through-wall cracks and that the FCG is insignificant for the 80-year evaluation period.

In addition, the licensee indicated that a generic FCG evaluation is not readily available for a typical loop bypass line. However, the licensee explained that the potential crack growth in the loop bypass line piping can be addressed by comparison to the plant-specific FCG analysis of the Surry SI piping, which is presented in Section 8.1.3, “Safety Injection Line FCG,” of WCAP-18491. As previously discussed, the FCG in the SI piping is insignificant and does not cause a through-wall cracking. The licensee’s rationale for why the FCG analysis of the SI piping is bounding for that of the loop bypass piping is further discussed below.

With respect to stresses, the licensee stated that the configuration of the loop bypass line precludes significant thermal stresses because (1) the loop bypass line valve is closed during normal plant operation, (2) the coolant temperature in the line can fluctuate only along with the respective temperatures of the hot-leg and cold-leg piping, and (3) accordingly, the piping segments between the RCL and the loop bypass line isolation valve will change very slowly only due to nominal turbulent penetration and buoyancy convection as transients occur in the RCL piping. Based on this evaluation, the licensee determined that these gradual and slow temperature variations would minimize the potential for fatigue. In addition, the licensee indicated the loop bypass line is 8-inch, Schedule 120 piping, which is larger and thicker than the 6-inch, Schedule 120 SI piping. The licensee explained that the smaller and thinner SI line piping conservatively bounds the loop bypass line with respect to mechanical stresses.

The NRC staff finds that the representative FCG evaluations for the pressurizer surge and the loop bypass 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 80 years of operation and would not cause through-wall cracks.

The licensee also addressed 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. Additionally, the licensee stated that field vibration measurements have been made on the RCL piping in a number of plants during hot functional testing. The stresses in the elbow below the RCP have been found analytically to be very small—between 2 and 3 thousand pounds per square inch (ksi) at the highest. 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. Additionally, vibrations in excess of what was assumed would be identified because during operation, an alarm signals the exceedance of the RCP shaft vibration limits.

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 (1) the cycles of the low-cycle fatigue meet the allowable cycle limit in accordance with the USAS B31.1 code, (2) the low-cycle fatigue in the RCS is also managed by the fatigue TLAA and fatigue monitoring program, (3) the FCG analyses for the subject piping support the conclusion that the potential FCGs are insignificant and do not affect crack stability and LBB applicability, and (4) the high-cycle fatigue due to pump vibration is insignificant.

3.2.2.2 Thermal Stratification

Section 2.4, “Other Possible Degradation During Service of the Auxiliary Piping Systems,” of WCAP-18491 states that thermal stratification occurs when conditions permit hot and cold layers of water to exist simultaneously in a horizontal pipe. This can result in significant thermal loadings due to the high fluid temperature differentials. Therefore, changes in the stratification state result in thermal cycling, which can cause fatigue degradation. The licensee also stated that the effects of stratification have been evaluated for the pressurizer surge lines, and these loads are used in the surge line LBB analysis of the LAR.

The licensee also performed a flaw tolerance evaluation on the pressurizer surge line in accordance with ASME Code, Section XI, Appendix L, for 60 years of operation, as approved by the NRC staff (ADAMS Accession No. ML18166A329). In conjunction with the flaw tolerance analysis using a postulated flaw, the licensee performs periodic inspections on the pressurizer surge line to confirm the structural integrity of the piping and the absence of piping degradation. In addition, the licensee will continue to use the Appendix L analysis and periodic inspections to manage cracking due to fatigue for the pressurizer surge line for the subsequent period of extended operation (60–80 years of operation), as approved by the NRC staff (ADAMS Accession No. ML20052F523).

In addition, the licensee’s inservice inspections have confirmed the absence of cracks due to thermal stratification. Therefore, the NRC staff finds that potential cracking due to thermal stratification is not an active degradation mechanism.

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 degrees F. The operating temperatures of the subject

piping lines are higher than 70 degrees 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 subject piping lines is approximately 656 degrees F, 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 800 degrees F, the temperature that would cause creep degradation in stainless steel material. Therefore, the NRC staff finds that creep is not an active degradation mechanism for the subject piping.

3.2.5 Wall Thinning

The licensee stated that wall thinning by erosion and erosion-corrosion should not occur in the accumulator, the RHR, the SI, the pressurizer surge, and the RCL bypass piping lines because the stainless steel material used for the piping is highly resistant to these degradation mechanisms. The licensee explained that as discussed in NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," issued September 1980 (ADAMS Accession No. ML070040195), a study on pipe cracking in PWR piping reported only two incidents of wall thinning in stainless steel pipe, and those were not in the accumulator, RHR, SI, pressurizer surge, or RCL bypass lines.

The NRC staff finds that wall thinning is not an active degradation mechanism for the subject piping because of the low velocity flow in the subject piping and the piping fabrication material that is resistant to wall thinning.

3.2.6 Water Hammer

The licensee indicates that there is a low potential for water hammer in the RCS and connecting auxiliary piping systems since they are designed and operated to preclude the voiding condition in the water-filled lines. The licensee explained that the design requirements for the subject piping lines are conservative relative to both the number of transients and their severity. In addition, the system design also considered relief valve actuation and the associated hydraulic transients following valve opening. To ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within a narrow range by the control rod positions, and pressure is also controlled within a narrow range for steady-state conditions by the pressurizer heaters and pressurizer spray. Accordingly, the flow characteristics of the RCS remain constant during a fuel cycle, which minimizes the potential for water hammer.

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-filled pipes during normal operation.

3.2.7 Conclusion on the Screening Based on Applicable Degradation Mechanisms

On the basis of the above evaluation, the NRC staff finds that the accumulator, the RHR, the SI, the pressurizer surge, and the RCL bypass lines in the scope of the LAR are not subject to any active degradation that can be a potential source of pipe rupture, consistent with SRP Section 3.6.3. The licensee also reviewed inservice inspection results and confirmed that there are no known relevant indications in the subject piping. Based on the absence of active degradation that can cause pipe rupture in the subject piping, the NRC staff concludes that the LBB analysis meets the acceptance criteria for a fracture mechanics analysis to be further used for the determination of crack stability in accordance with SRP Section 3.6.3.

3.3 Fracture Mechanics 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 section also specifies that the licensee should provide the material properties, including toughness, tensile data, and long-term effects such as thermal aging.

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) and A403 WP316 stainless steel (wrought fittings). 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 licensee further indicated that the weld processes used in the subject piping are submerged arc weld (SAW) and shielded metal arc weld (SMAW) processes.

The licensee's LBB analysis used ASME Code mechanical properties to establish the tensile properties of the piping materials, consistent with ASME Code, Section II, 2007 Edition. The material modulus of elasticity was also interpolated from ASME Code values for the operating temperatures considered, and Poisson's ratio was taken as 0.3. Section 4.0, "Material Characterization," of WCAP-18491 provides the yield strengths, ultimate strengths, and elastic moduli for the materials of each subject piping.

The NRC staff finds the licensee's approach to be acceptable because (1) the licensee adequately used the code property data to estimate the operating temperature properties of the subject piping, and (2) the licensee accounted for the temperature effects on material properties by interpolating material property data at different temperatures.

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 WCAP-18491 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 licensee used the combinations of the normal and faulted loading conditions (also called loading case combinations) to estimate leakage crack sizes and critical crack sizes. The NRC staff notes that these loading case combinations adequately account for thermal stratification loads and represent the normal and faulted loading conditions, because the licensee considered the bounding loads of the relevant transients that can cause thermal stratification and included both normal and faulted conditions in the load analysis.

The NRC staff finds that the licensee's load combinations in the crack stability analysis of the subject piping are acceptable because (1) the licensee used the absolute sum load combination method, (2) the licensee appropriately considered the upper bound loads under the faulted conditions in the limiting load combinations (including deadweight, thermal expansion, safe shutdown earthquake, and seismic anchor motion loads), (3) the calculations of the total moment considered the bending and torsional moments, (4) thermal stratification stresses are considered in the loading case combinations for the normal operation and faulted conditions of the pressurizer surge lines, and (5) 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 system can detect.

The licensee stated that the RCS pressure boundary leakage detection system can detect a leak rate of 1 gpm. Accordingly, the LBB analysis for the subject piping lines uses a leakage rate of 10 gpm, which is 10 times the leak rate detection capability. Section 3.4 of this safety evaluation documents the NRC staff's evaluation of the leakage detection capability.
The licensee explained that the analysis of single-phase flow (e.g., low temperature flow) considers the frictional pressure losses, including the flow passage, inlet, and outlet losses. The licensee also 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 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," issued September 1983 (ADAMS Accession No. ML20078E026).

Based on its review of the licensee's leakage crack size calculation method, the NRC staff finds that the estimated size (length) of the leakage crack is large enough 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 the given leakage detection limit are consistent with those used in the existing LBB analysis for the primary coolant loops, which constitute the current licensing basis of Surry Units 1 and 2 (see WCAP-15550-NP, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Surry Units 1 and 2 Nuclear Power Plants for the Subsequent License Renewal Program (80 Years) Leak-Before-Break Evaluation," Revision 2, issued March 2019 (ADAMS Accession No. ML19095A605)).

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 limiting locations were established on the basis of the pipe geometry, welding process, material type, operating temperature, operating pressure, and the highest faulted stresses at the welds. Therefore, the critical locations are established based on the bounding stresses and piping material properties.

As part of the crack stability analysis, the licensee 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 (including internal pressure), pipe dimensions, and crack size and configuration. For the limit load method, the licensee also multiplied the pipe loads by the Z factor, considering the welding processes (i.e., SMAW and SAW processes). The licensee stated that it derived the Z factors for SMAW and SAW in accordance with SRP Section 3.6.3.

For piping segment SI-HL-I, the licensee performed a J-integral fracture mechanics analysis and demonstrated that the applied J-integral value is less than the fracture toughness (J_{IC}) of the piping base material. Accordingly, the postulated through-wall crack, which has a crack size twice the leakage crack size, would not result in crack tip extension.

The NRC staff finds the crack stability analysis to be acceptable because (1) a safety margin of 2 is demonstrated between the leakage crack size and the critical crack size, and (2) the licensee's methodology is consistent with the guidance in SRP Section 3.6.3.

3.3.5 Power Uprate

In a previous licensing action, the licensee implemented a measurement uncertainty recapture (MUR) power uprate that resulted in 1.6-percent thermal power increase (ADAMS Accession Nos. ML100320264 and ML101750002). The licensee explained that the engineering evaluations performed for the MUR program concluded there was no significant impact to the RCL and branch line piping analysis as a result of the small temperature increase associated with the power uprate. The NRC staff finds the licensee's evaluation of the MUR uprate conditions to be acceptable because the licensee clarified that the MUR power uprate caused negligible effects on the operating and loading conditions of the subject piping.

3.3.6 License Renewal

In relation to the FCG analysis discussed in Section 3.2.2.1 of this safety evaluation, 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 indicated that the FCG analyses for the subject piping bound the transient projections for 80 years of operation. The NRC staff finds the licensee's evaluation to be acceptable because the licensee has performed FCG analyses, which involve the time dependencies of crack growth, and confirmed the validity of the LBB analysis for the subsequent period of extended operation (i.e. 80 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 described in Electric Power Research Institute NP-3596-SR, "PICEP: Pipe Crack Evaluation Program," Revision 1, issued December 1987. The analysis evaluated the following critical locations described in Table 6-1, "Flaw Sizes Yielding a Leak Rate of 10 gpm for the Surry Units 1 and 2 Pressurizer Surge Lines," of WCAP-18491: (1) Node 10, (2) Node 18, and (3) Node 250. The PICEP code conducts a limit load analysis with Z factors applied to the loads, which are used in the licensee's evaluation. In the analysis, the NRC staff confirmed that the critical crack sizes determined by the licensee are in 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 critical crack sizes.

The NRC staff also used the PICEP code to estimate the leakage crack sizes and to check the margin between the leakage crack size and the critical crack size. The confirmatory analysis used the same leakage rates as those used in the licensee's analysis (10 gpm). In the

confirmatory analysis, the NRC staff confirmed that a margin of 2 exists between the leakage crack size and the critical crack size.

3.4 Reactor Coolant Pressure Boundary Leakage Detection System Capability

The NRC staff has previously evaluated the capability of Surry RCPB leakage detection system and determined it was acceptable. Surry UFSAR Section 14.5.1.2, "Description of Events," states the following:

> A revision to General Design Criterion 4 (GDC-4) was issued by the NRC effective May 12, 1986. In accordance with the revised rule, consideration of the dynamic effects of RCS pipe rupture may be eliminated as a design basis provided the "Leak Before Break" (LBB) analyses demonstrate that any flaw in the RCS primary loop piping which grew would become a through-wall crack with detectable leakage allowing shutdown of the plant long before a rupture would occur. LBB fracture mechanics analyses applicable to Surry have been accepted by the NRC and, in accordance with Amendment 108 to the Surry operating license, consideration of the dynamic effects of a LOCA is no longer part of the design basis.

Surry UFSAR Section 4.2.7.1, "Leakage Detection," states that coolant leakage from the RCS to the containment is indicated in the control room by one or more of the following methods:

1. The containment air particulate monitoring system—A system is provided to monitor particulate activity from the areas enclosing the reactor coolant system components so that any leakage from them can be easily detected. The containment air particulate monitor is indicated, recorded, and alarmed in the control room.

2. The containment gas monitor—A system is provided to monitor gaseous activity from areas enclosing the reactor coolant system. Even though the gas monitor itself is less sensitive than the particulate monitor, the gaseous activity from any leakage is expected to be higher than the particulate activity, so that the gas monitor will also be sensitive to a leak. The containment gas monitor is indicated, recorded, and alarmed in the control room.

3. Abnormal makeup water requirements—Any leakage will cause an increase in the amount of makeup water required to maintain normal level in the pressurizer. The primary-grade water and concentrated boric acid makeup flow rate are both recorded and alarmed in the control room.

4. Containment instrumentation—The reactor containment sump level instrumentation and the containment pressure and temperature instrumentation could all indicate leakage in the containment, but not necessarily from the primary coolant system. These measurements are also subject to variations unrelated to leakage from ruptured fluid systems. The instruments all indicate in the control room; however, it is not the primary purpose of the containment sump level and containment pressure or temperature instrumentation to detect primary coolant system leakage. Primary coolant system leakage can be most readily detected by increased makeup requirements to the primary coolant system and by the

containment gas monitors. The containment pressure and temperature are recorded by the data logger. The containment pressure also alarms in the control room.

5. Reactor vessel leakoff—Leakage through the reactor vessel head flange will leak off between the double o-ring seal to the leakoff provided. Leakage into this leakoff will cause high temperature in this line, which will actuate an alarm in the control room.

UFSAR Section 4.2.7.1 also states that methods 1 and 2 can only be used for leakage detection if there are enough activated products in the reactor coolant. If no such activated products are in the reactor coolant, the other methods can be used to detect a leak.

The NRC staff finds that the licensee has sufficiently described the performance of the leakage detection systems in Surry UFSAR Section 4.2.7.1 under various plant conditions, including conditions where RCS activity would be low due to fuel cladding integrity.

UFSAR Section 4.2.7.1 states that in accordance with the information provided in a letter from Virginia Electric and Power Company to the NRC, “Request for Partial Exemption from General Design Criterion 4—Supplement,” dated December 3, 1985 (ADAMS Package Accession No. ML18144A041), and the safety evaluation of Surry Units 1 and 2, License Amendments No. 108, dated June 16, 1986 (ADAMS Accession No. ML012830465), “it was concluded that the reactor coolant leakage detection capability meets the staff guidelines of 1 gpm in 4 hours.”

The Virginia Electric and Power Company letter, which provided supporting basis for License Amendments No. 108, addressed all regulatory positions of RG 1.45, Revision 0, issued May 1973, providing design guidelines for RCPB leakage detection and collection systems. Design Guideline 5 of RG 1.45, Revision 0 states that,“[t]he sensitivity and response time of each leakage detection system in regulatory position 3 above employed for unidentified leakage should be adequate to detect a leakage rate, or its equivalent, of 1 gpm in less than 1 hour.” With respect to this, the Virginia Electric and Power Company letter stated the following:

> Generic Letter 84-04 allows an exception to the Regulatory Guide 1.45 by requiring "at least one leakage detection system with a sensitivity capable of detecting 1 gpm in 4 hours must be operable." The sensitivity and response time of the methods described for detection of RCS leakage are dependent on RCS activity, previous leakage rate which can increase steady state count rates on the gross and particulate radiation monitoring system, frequency of surveillance, etc. Different detection methods would be effective over a broad range of operating conditions. Given the depth and redundancy of leakage detection methods described above, we are confident a 1 gpm leak rate would be detected within 4 hours during steady state operation.

The NRC staff stated in its safety evaluation of License Amendments No. 108, dated June 16, 1986, that “the staff concludes that leakage detection capability at Surry Power Station, Units 1 and 2 meets the staff guidelines of 1 gpm in 4 hours, as stated in Generic Letter 84-04,” (ADAMS Accession No. ML012830465).

The licensee stated the following in Section 3.0 of LAR Attachment 1:

> Through-wall flaw sizes were postulated which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was assessed and shown not to be an issue for the pressurizer surge, RHR, SI accumulator, loop bypass and SI piping connected to the RCLs.

As discussed above, the NRC staff has previously reviewed and accepted the RCPB leak detection system for Surry Units 1 and 2 and the licensee is not proposing any changes to TS or leakage detection methods. The NRC staff finds that the leak detection system for Surry Units 1 and 2 meets the intent of RG 1.45, Revision 0, and meets a leak detection capability of 1.0 gpm, which provides a margin of 10 to the evaluated crack leakage rate, and, therefore, it is acceptable.

3.5 Technical Conclusion

On the basis of its review of the LAR, as supplemented, the NRC staff finds that, for the subject accumulator, the RHR, the SI, the pressurizer surge, and the RCL bypass line piping, the licensee has demonstrated that (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 and fracture mechanics analyses, (5) the licensee's LBB methods are consistent with the guidance in SRP Section 3.6.3, and (6) the potential FCG in the subject piping is insignificant and does not affect the crack stability and the validity of the LBB analysis.

Accordingly, the NRC staff finds that the licensee's analysis has demonstrated that the subject piping has an extremely low probability of rupture. As described in Section 3.1 of this safety evaluation, the LBB application pertains to specific RCS piping segments associated with the accumulator, the RHR, the SI, the pressurizer surge, and the RCL bypass lines.

Based on the evaluation above, the NRC staff has concluded that the Surry LBB leakage detection remains consistent with (1) the requirement of 10 CFR 50.36(c)(2)(i), (2) the requirement of 10 CFR Part 50, Appendix A, GDC 30, and (3) the intent of RG 1.45, Revision 0, for RCS leakage detection capabilities. The NRC staff also concludes that the use of the current TS leakage limit for the proposed LBB application is acceptable because (1) the TS requires that, if the leakage rate exceeds 1 gpm and the source of leakage cannot be identified in 4 hours, the reactor must be brought to shutdown, and (2) the leakage detection system is capable of detecting 1 gpm leakage, consistent with the leakage rate used in the LBB analysis.

Pursuant to 10 CFR Part 50, Appendix A, GDC 4, 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, the RHR, the SI, the pressurizer surge, and the RCL bypass line piping from the current licensing basis at Surry Units 1 and 2.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC notified an official from the Virginia Division of Radiological Health of the proposed issuance of the amendment. On June 15, 2021, the Virginia State official confirmed that the Commonwealth of Virginia 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, "Standards for protection against radiation." 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 off site, 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* (86 FR 7118) on January 26, 2021, 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). Under 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to 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 public health and safety 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 public health and safety.

Principal Contributors: S. Min
H. Wagage

Date: August 20, 2021

SUBJECT: SURRY POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 304 AND 304 RE: LEAK-BEFORE-BREAK FOR PRESSURIZER SURGE, RESIDUAL HEAT REMOVAL, SAFETY INJECTION ACCUMULATOR, REACTOR COOLANT SYSTEM BYPASS AND SAFETY INJECTION LINES (EPID L-2020-LLA-0255) DATED AUGUST 20, 2021

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