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

February 6, 2020

Mr. John A. Krakuszeski
Vice President
Brunswick Steam Electric Plant
Duke Energy Progress, LLC
8470 River Rd. SE (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 298 AND 326 TO MODIFY TECHNICAL SPECIFICATIONS TO EXTEND CONTAINMENT LEAKAGE TEST INTERVALS (EPID L-2019-LLA-0031)

Dear Mr. Krakuszeski:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 298 to Renewed Facility Operating License No. DPR-71 and Amendment No. 326 to Renewed Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2. The amendments are in response to your application dated February 27, 2019, as supplemented by letter dated July 25, 2019.

The amendments modify Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program." This change extends the maximum interval for the Integrated Leakage Rate Test from 10 years to 15 years and the maximum interval for the containment isolation valve local leak rate tests from 60 months to 75 months.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* Notice.

Sincerely,

/RA/

Andrew Hon, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 298 to License No. DPR-71
2. Amendment No. 326 to License No. DPR-62
3. Safety Evaluation

cc: Listserv

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 298
Renewed License No. DPR-71

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

   A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated February 27, 2019, and supplemented on July 25, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 298, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: February 6, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 298

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace page 6 of Renewed Facility Operating License No. DPR-71 with the attached page 6.

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

Remove Pages	Insert Pages
5.0-15	5.0-15
5.0-16	5.0-16

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180^{th} day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

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

(1) Maximum Power Level
The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts thermal.

(2) Technical Specifications
The Technical Specifications contained in Appendix A, as revised through Amendment No. 298, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 203 to Renewed Facility Operating License DPR-71, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 203. For SRs that existed prior to Amendment 203, including SRs with modified acceptance criteria and SRs whose frequency of

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A primary containment leakage rate testing program shall establish requirements to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the Limitations and Conditions specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exceptions:

a. The visual examination of concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

b. The visual examination of the metallic shell, penetrations, and appurtenances intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

c. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 3-A;

d. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

e. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

(continued)

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

f. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-2002.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each air lock door, leakage rate is ≤ 5 scfh when the gap between the door seals is pressurized to ≥ 10 psig.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program frequencies.

5.5.13 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation (CREV) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:

(continued)

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 326
Renewed License No. DPR-62

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

   A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated February 27, 2019 and supplemented on July 25, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 326, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: February 6, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 326

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace page 6 of Renewed Facility Operating License No. DPR-62 with the attached page 6.

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

Remove Pages	Insert Pages
5.0-15	5.0-15
5.0-16	5.0-16

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except Item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

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

(1) Maximum Power Level
The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts (thermal).

(2) Technical Specifications
The Technical Specifications contained in Appendix A, as revised through Amendment No. 326, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 233 to Renewed Facility Operating License DPR-62, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 233. For SRs that existed prior to Amendment 233, 5.5.11 Safety Function Determination Program (SFDP) (continued)

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A primary containment leakage rate testing program shall establish requirements to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the Limitations and Conditions specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exceptions:

a. The visual examination of concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

b. The visual examination of the metallic shell, penetrations, and appurtenances intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

c. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 3-A;

d. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

e. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

(continued)

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

f. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-2002.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each air lock door, leakage rate is ≤ 5 scfh when the gap between the door seals is pressurized to ≥ 10 psig.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program frequencies.

5.5.13 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation (CREV) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 298

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 326

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-62

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By application dated February 27, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19058A768) as supplemented by letter dated July 25, 2019 (Accession No. ML19206B231), Duke Energy Progress, LLC (Duke Energy or the licensee) submitted a License amendment request (LAR) for Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The LAR would revise Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to adopt Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors* (Accession No. ML12221A202)," and the limitations and conditions specified in NEI 94-01, Revision 2-A (Accession No. ML100620847), as the guidance document for implementation of the performance-based Option B of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J. The revision would allow the maximum interval for the Integrated Leakage Rate Test (ILRT) to extend from 10 years to 15 years and the maximum interval for the containment isolation valve (CIV) local leak rate tests (LLRT) to extend from 60 months to 75 months. In the TS, the references to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (Accession No. ML003740058) would be deleted, the references to NEI 94-01, Revision 0, would be changed to NEI 94-01, Revision 3-A, and the references to American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-1994 would be changed to ANSI/ANS 56.8-2002.

The July 25, 2019, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 7, 2019 (84 FR 19967).

2.0 REGULATORY EVALUATION

BSEP Units 1 and 2 are General Electric boiling water reactors with Mark I water pool pressure suppression design primary containments. These primary containments consist of reinforced concrete pressure vessels with leak-tight steel plate liners on all inside surfaces. A drywell volume is composed of vertical right cylinders and truncated cones with an inside diameter varying from about 36 feet to about 65 feet in a configuration similar to an inverted lightbulb. The height from the foundation mat to the drywell head flange is about 111 feet. A 35-foot 10-inch steel domed removable steel head is bolted to a liner extension at the top of the reinforced concrete portion of the drywell. The suppression chamber is a hollow reinforced concrete shell of rectangular cross-section encircling the lower portion of the drywell containment structure. The concrete encloses 16 continuous cylindrical sections of a 3/8-inch thick torus steel liner. The cross-sectional diameter of the suppression chamber is 29 feet and the major centerline of the torus is 109 feet.

The suppression chamber is structurally independent of the drywell and is filled approximately half-way with water. The drywell is mostly above and centered over the wetwell and these volumes are connected by large vent pipes. The drywell houses the reactor vessel, the recirculation loops, main steam lines and other branch connections of the reactor coolant system. This containment arrangement limits primary containment pressurization from a design base loss of coolant accident (LOCA) by channeling steam and heated drywell atmosphere into the suppression pool where condensation and cooling of the containment atmosphere occurs during initial blowdown. The suppression pool also serves as a water source for emergency core cooling and containment spray pumps.

The primary containment provides a leak tight barrier against the potential uncontrolled release of fission products during a LOCA. TS 5.5.12 identifies the primary containment allowable leakage rate (L_a) as 0.5 percent of the containment air weight per day at the calculated maximum LOCA pressure (P_a) of 49 pound square inch gauge (psig).

<u>Licensee's Proposed Changes</u>

BSEP Units 1 and 2 TS 5.5.12 currently state, in part:

> A primary containment leakage rate testing program shall establish requirements to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995, as modified by the following exceptions:
>
> a. The visual examination of concrete surfaces intended to fulfill the requirements of 10 CFR Part 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.
>
> b. The visual examination of the metallic shell, penetrations, and appurtenances intended to fulfill the requirements of 10 CFR Part 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by

the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

c. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at Pa as specified in Nuclear Energy Institute Guideline 94-01, Revision 0;

d. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

e. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

f. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than Pa instead of leak rate testing at Pa as specified in ANSI/ANS 56.8-1994.

With the proposed changes, BSEP Units 1 and 2 TS 5.5.12 would state in part:

A primary containment leakage rate testing program shall establish requirements to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the Limitations and Conditions specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exceptions:

a. The visual examination of concrete surfaces intended to fulfill the requirements of 10 CFR Part 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

b. The visual examination of the metallic shell, penetrations, and appurtenances intended to fulfill the requirements of 10 CFR Part 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

c. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at Pa as specified in Nuclear Energy Institute Guideline 94-01, Revision 3-A;

d. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

e. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

f. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than Pa instead of leak rate testing at Pa as specified in ANSI/ANS 56.8-2002.

With this change, BSEP Units 1 and 2 will implement the guidance of NEI 94-01, Revision 3-A and the conditions and limitations of NEI 94-01, Revision 2-A rather than those of NEI 94-01, Revision 0 that was endorsed by RG 1.163, Revision 0. NEI 94-01, Revision 3-A provides that extension of the Type A test (ILRT) interval to 15 years be based on two consecutive successful Type A tests (performance history) and other requirements stated in Section 9.2.3 of NEI 94-01. The basis for acceptability of extending the Type A test interval also includes implementation of robust Type B and Type C testing of the penetration barriers through which most containment leakage has historically been shown to occur, and that are expected to continue to be pathways for primary containment leakage. The basis also includes a robust containment visual inspection program where deterioration of the primary containment boundary away from penetrations can be detected and remediated. NEI 94-01, Revision 3-A also provides that Type C test intervals may be extended to 75 months based on two consecutive successful tests (performance history) and meeting other specified conditions.

The existing BSEP TS 5.5.12 identify exceptions to NEI 94-01 guidance. The revised TS 5.5.12 will contain similar exceptions to the corresponding portions of the NEI 94-01 Revision 3-A and ANSI/ANS 56.8 – 2002.

Under 10 CFR 50.54(o), primary reactor containments for water cooled power reactors are subjected to the requirements set forth in Appendix J to 10 CFR Part 50, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.” Appendix J of 10 CFR Part 50, must be met for the amendment to be granted. It includes two options: “Option A – Prescriptive Requirements,” and “Option B – Performance-Based Requirements,” either of which may be chosen by a licensee for meeting the requirements of Appendix J. BSEP Units 1 and 2 adopted 10 CFR Part 50, Appendix J, Option B for Type A (Integrated leakage rate test (ILRT)), and Type B and Type C (Local leakage rate test (LLRT)), respectively, by license amendments 181 and 213 dated February 1, 1996.

The testing requirements in 10 CFR Part 50, Appendix J ensure that (a) leakage through containments or systems and components penetrating containments do not exceed allowable leakage rates specified in the TS; and (b) integrity of the containment structure is maintained during the service life of the containment.

Appendix J, Option B of 10 CFR Part 50, specifies performance-based requirements and criteria for preoperational and subsequent leakage rate testing. These requirements are met by performing:

- Type A tests to measure the containment system overall integrated leakage rate;

- Type B pneumatic tests to detect and measure local leakage rates across pressure-retaining leakage-limiting boundaries such as penetrations; and

- Type C pneumatic tests to measure CIV leakage rates.

After the preoperational tests, these tests are required to be conducted at periodic intervals based on the historical performance of the overall containment system (for Type A tests), and based on the safety significance and historical performance of each penetration boundary and isolation valve (for Type B and C tests), to ensure integrity of the overall containment system as a barrier to fission product release. The leakage rate test results must not exceed the L_a with margin, as specified in the TS. Option B also requires a general visual inspection of the accessible interior and exterior surfaces of the containment system prior to each Type A test and at a periodic interval between tests, to identify structural deterioration which may affect the containment leak-tight integrity.

Section V.B.3 of 10 CFR Part 50, Appendix J, Option B, requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage-testing program must be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the NRC staff and endorsed in a regulatory guide. The implementation document that is currently referenced in the BSEP Unit Nos. 1 and 2, TS 5.5.12, "Primary Containment Leakage Rate Testing Program," is RG 1.163, “Performance-Based Containment Leak Test Program,” dated September 1995. RG 1.163 endorsed TR NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995, as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B to Appendix J to 10 CFR Part 50, subject to four regulatory positions delineated in Section C of the RG. NEI 94-01, Revision 0, includes provisions that allow the performance-based Type A test interval to be extended to up to 10 years, based upon two consecutive successful tests.

NEI 94-01, Revisions 2-A and 3-A, have been reviewed by the NRC and approved for use in LARs. The final Safety Evaluation (SE) for Revision 2, issued by letter dated June 25, 2008 (ADAMS Accession No. ML081140105), documents the NRC’s evaluation and acceptance of Revision 2, subject to six specific limitations and conditions listed in Section 4.1 of the NEI 94-01 Revision 2 SE. The final SE for Revision 3, issued by letter dated June 8, 2012 (ADAMS Accession No. ML121030286), includes two specific limitations and conditions listed in Section 4.0 of that SE for the Type C tests. In addition, Section 9.2.3.4, “Plant-Specific Confirmatory Analyses,” of NEI 94-01 states that “the assessment should be performed using the approach and methodology described in Electric Power Research Institute (EPRI) Technical Report (TR) 1018243[1], “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals.” The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.”

3.0 TECHNICAL EVALUATION

The licensee justified in the LAR, and the NRC staff reviewed the proposed TS changes in the following three aspects consistent with the guidance in NEI 94-01, Revision 2-A and 3-A:

1. Historical plant-specific containment leakage testing program results,

2. Containment In-Service Inspection (CISI) program results, and

3. Supporting plant-specific risk impact assessment of extended integrated leak rate testing intervals

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[1] EPRI TR-1018243, is also identified as EPRI TR-1009325, Revision 2-A. This report is publicly available and can be found at www.epri.com by typing “1018243” in the search field box.

3.1 Historical Plant-Specific Containment Leakage Testing Results Evaluation

3.1.1 Historical Type A Test (ILRT) Results

In LAR Table 3.4.4-1, BSEP Unit 1 Type A ILRT Testing History, the licensee presented the historical results of the Unit 1 tests as summarized below.

Test Date	Leakage Rate (Identified as As-Found in the LAR Table) (Primary Containment Atmosphere Percent Weight per Day)	Leakage Rate Performance Criterion (Primary Containment Atmosphere Weight Percent per Day)
03/25/2004	0.3351	0.5
04/11/2010	0.201	0.5

The ILRT results back to 1976 were provided and give a general historical context for the typical range of leakage values determined. With the adoption of Appendix J, Option B, how the ILRT result was determined has changed and the two most recent test results are summarized here as the requirement for being able to extend the ILRT interval out to 15 years (i.e., based largely on the last two periodic tests having met the performance criterion). Earlier test results are generally not directly comparable to those obtained later as one or more of the following have likely changed: test methods, test pressure, allowable leakage criteria and consequence dose analyses separation of leakages into pathways with differing airborne radioactivity attenuation characteristics. The NEI 94-01, Revision 3-A, requirement for allowing the extended test interval is that the past two tests meet the performance criterion by showing a leakage of L_a or less. Both the 2004 and 2010 results show considerable margin to the performance criterion. These results suggest that an ILRT interval of 15 years would not result in exceeding the performance criterion for Unit-1.

In LAR Table 3.4.4-2, BSEP Unit 2 Type A ILRT Test History, the licensee presented the historical results of the Unit 2 tests as summarized below.

Test Date	Leakage Rate (Identified as As-Found in the LAR Table) (Primary Containment Atmosphere Percent Weight per Day)	Leakage Rate (Identified as As-Left in the LAR Table) (Primary Containment Atmosphere Percent Weight per Day)	Leakage Rate Performance Criterion (Primary Containment Atmosphere Weight Percent per Day)
03/30/2005	0.4255	0.389	0.5
04/01/2015	0.4797	0.299	0.5

The ILRT results back to 1974 were provided and give a general historical context for the typical range of leakage values determined. With the adoption of Appendix J, Option B, how the ILRT result was determined has changed. The two most recent test results are summarized here because the justification for being able to extend the ILRT interval out to 15 years (i.e., based largely on the last two periodic tests having met the performance criterion). Earlier test results are not directly comparable to those obtained later as one or more of the following have likely changed: test methods, test pressure, allowable leakage criteria and consequence dose analyses separation of leakages into pathways with differing airborne radioactivity attenuation characteristics. The NEI 94-01, Revision 3-A, requirement for allowing the extended test interval is that the past two tests meet the performance criterion by showing a leakage of L_a or less. Both the 2005 and 2015 result show margin to the performance criterion. With Option B the intent is that there normally be one ILRT performance recorded if there is no corrective maintenance needed to the containment boundary away from the penetrations that could affect containment leakage, no as-found ILRT result and no as-left ILRT result as there was with Option A. This intent was emphasized in Condition 4.1.1 of the SE for NEI 94-01, Revision 2, that the scheme of determining an as-found ILRT and as-left ILRT by factoring in or out "leakage savings" was no longer needed. If this had been used for the last 2 ILRTs, then the ILRT results would have been closer to or the same as what the licensee shows as the as-left ILRT value, and that considerable margin to the performance criterion L_a is apparent. The licensee may but is not obligated to recalculate results to current methods or definitions for comparison and determination of having passed the ILRT. These results suggest that an ILRT interval of 15 years would not result in exceeding the performance criterion for Unit-2 if the method of NEI 94-01, Revision 2/3-A is used as emphasized by Condition 4.1.1 of the SE for NEI 94-01, Revision 2.

3.1.2 Historical Type B and Type C Test Results Combined Totals

In LAR Table 3.6.5-1 the licensee presented the historical results of the Type B and C test combined as-found minimum pathway leakage totals for BSEP Unit 1 as summarized below:

Refuel Outage / Year	As-Found Minimum Pathway Leakage Rate (standard cubic feet per hour) (scfh)	Percent of TS 5.5.12 Combined Type B and C Performance Criterion with Margin (0.6 L_a)
B117R1 / 2008	54.397	34.04
B118R1 / 2010	64.885	40.61
B119R1 / 2012	124.662	78.02
B120R1 / 2014	40.734	25.49
B121R1 / 2016	35.013	21.91
B122R2 / 2018	29.474	18.45

The performance criterion for combined Type B and C test total summed from the as-found minimum pathway leakage values with the TS 5.5.12 specified margin is 0.6 L_a. The licensee stated in LAR Section 3.6.5 that 0.6 L_a is 159.78 standard cubic feet per hour (scfh) for each Unit. These combined Type B and C test totals show sufficient additional margin to suggest that the performance criteria are unlikely to be exceeded by allowing BSEP Unit 1 ILRT maximum interval to be extended to 15 years or the Type C test intervals to be extended to 75 months.

In LAR Table 3.6.5-2 the licensee presented the historical results of the Type B and C test combined as-found minimum pathway leakage totals for BSEP Unit 2 as summarized below:

Refuel Outage / Year	As-Found Minimum Pathway Leakage Rate (standard cubic feet per hour) (scfh)	Percent of TS 5.5.12 Combined Type B and C Performance Criterion with Margin (0.6 L_a)
B218R1 / 2007	64.108	40.12
B219R1 / 2009	36.255	22.69
B220R1 / 2011	51.365	32.15
B221R1 / 2013	56.465	35.34
B222R1 / 2015	54.659	34.21
B223R2 / 2017	35.986	22.52

These combined Type B and C test totals show sufficient additional margin to suggest that the performance criteria are unlikely to be exceeded by allowing BSEP Unit 2 ILRT maximum interval to be extended to 15 years or the Type C test intervals to be extended to 75 months.

3.1.3 NRC SE Conditions in NEI 94-01, Revision 2-A

In the NRC SE dated June 25, 2008 (ADAMS Accession No. ML100620847), the NRC staff concluded that the guidance in Topical Report, NEI 94-01, Revision 2, is acceptable for reference by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to six conditions. The requirements of NEI 94-01 stayed essentially the same from the original version through Revision 2 except that the regulatory positions of RG 1.163 were incorporated and the maximum ILRT interval extended to 15 years. The LAR Table 3.9.1 described the licensee response to the six conditions identified in the SE dated June 25, 2008, and the NRC staff evaluated these responses to determine whether the licensee adequately addressed these conditions.

a. NRC Condition 1

"For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS- 56.8-2002. (Refer to SE Section 3.1.1.1)"

The licensee stated in the LAR that they would be using the definition in NEI 94-01, Revision 3-A, Section 5.0, which remained the same from NEI 94-01, Revision 2-A. This definition is the one identified as acceptable and, therefore, the licensee has addressed and satisfied NRC Condition 1.

b. NRC Condition 2

"The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3)"

The LAR provided the historical performance dates and current schedule of containment inspections in the LAR Section 3.6.3 and 3.6.4 for the BSEP Units 1 and 2 third containment class MC/CC components inservice inspection intervals by period for implementation of ASME Boiler & Pressure Vessel Code Section XI, Subsections IWE, Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants, and Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Plants. Therefore, the licensee addressed and satisfied NRC Condition 2.

c. NRC Condition 3

"The licensee addresses the areas of the containment structure potentially subjected to degradation."

The licensee described in LAR Sections 3.6.3 and 3.6.4 and elsewhere identified:

- The containment structures that are potentially subjected to degradation,
- currently no unacceptable degradation regarding degradation found in past inspections in accordance with the ASME Code, Section XI, and
- how inaccessible areas would be assessed for degradation potential.

Therefore, the licensee addressed and satisfied NRC Condition 3.

d. NRC Condition 4

"The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4)"

The LAR indicated that no major modifications of the containment are planned and that there have been no major modifications of the containment since the last ILRTs. In addition, neither the containment nor the containment isolation system modifications were required at BSEP to comply with the NRC Orders for FLEX. Therefore, the licensee has addressed and satisfied NRC Condition 4.

e. NRC Condition 5

"The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2)"

The licensee response in the LAR indicates acknowledgement and acceptance of this NRC staff position. Therefore, the licensee addressed and satisfied NRC Condition 5.

f. NRC Condition 6

"For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past ILRT data."

This condition is not applicable to BSEP Units 1 and 2 as they were not licensed under 10 CFR Part 52.

3.1.4 NRC SE Conditions in NEI 94-01, Revision 3-A

In the NRC Safety Evaluation Report dated June 8, 2012 (ADAMS Accession No. ML121030286), the NRC staff concluded that the guidance in TR NEI 94-01, Revision 3-A, is acceptable for reference by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to two conditions.

a. NRC Condition 1

NEI TR 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The NRC staff is allowing the extended interval for Type C LLRTs be increased to 75 months with the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The NRC staff is also allowing the non-routine emergent extension out to 84-months as applied to Type C valves at a site, with some exceptions that are detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g. BWR MSIVs), and those valves with a history of leakage, any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months.

b. NRC Condition 2

"When scheduling a valve LLRT interval beyond 60-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B and C combined totals and must be included in the post-outage report. The report must include the reasoning and determination of the acceptability of the extensions, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations."

The licensee indicated in the LAR that the BSEP post-outage Appendix J testing summary reports will include the margin between the Type B and Type C minimum pathway leak rate summation value adjusted for understatement and the acceptance criterion. Should the Type B and C combined totals exceed an administrative limit of 0.5 L_a but be less than the TS acceptance value (performance criterion) of 0.6 L_a, then an analysis will be performed, and a corrective action plan prepared to restore and maintain the leakage summation margin to less than the BSEP administrative limit. The LAR also stated that BSEP will apply the 9 months grace period only to eligible Type C tested components and only for non-routine emergent conditions. The licensee acknowledges these two conditions and the likelihood that longer test intervals would increase the understatement of actual leakage potential given the method by which the totals are calculated and will assign additional margin for monitoring acceptability of results via administrative limits and understatement contribution adjustments. Therefore, the licensee addressed and satisfied NRC Conditions 1 and 2 of NEI 94-01, Revision 3-A.

3.1.5 Historical Containment Leakage Testing Results Evaluation Summary

Based on the above historical leakage testing results evaluation, the NRC staff finds that the results of the recent ILRTs and of the LLRT (Type B and Type C tests) combined totals demonstrate acceptable performance and support a conclusion that the structural and leak-tight integrity of the primary containment vessel is adequately managed and will continue to be periodically monitored and managed effectively. The NRC staff finds that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting TR NEI 94-01, Revision 3-A, and the limitations and conditions identified in the NRC staff SE incorporated in TR NEI 94-01, Revision 2-A.

3.2 Containment In-Service Inspection Results Evaluation

In Reference 5.1, the licensee stated that the BSEP CICS Program is currently beginning its third inspection interval. The third inspection interval for BSEP became effective on May 11, 2018 and will end on May 10, 2028 and the governing code is the 2007 Edition with the 2008 Addenda of the ASME Code, Section XI. The containment plan is being merged into the overall ISI plan which is in its fifth interval beginning May of 2018 and ending May of 2028. Going forward for consistency, the Containment plan will be referenced by the licensee as being in the fifth inspection interval. The licensee states that the CICS inspection schedules are as follow:

Table 3.2 Inspection Schedule for BSEP Unit Nos. 1 and 2 Containments

<table>
<tr><th colspan="3">Unit 1 Containment MC/CC Inspection Interval[4]</th></tr>
<tr><td>First Period [2,3]</td><td>Second Period [2,3]</td><td>Third Period [2]</td></tr>
<tr><td>5/11/2018 – 5/10/2022</td><td>5/11/2022 – 5/10/2025</td><td>5/11/2025 – 5/10/2028[1]</td></tr>
<tr><td>Outage 1 (B1R23)</td><td rowspan="2">Outage 3 (B1R25)</td><td>Outage 4 (B1R26)</td></tr>
<tr><td>Outage 2 (B1R24)</td><td>Outage 4 (B1R27)</td></tr>
<tr><th colspan="3">Unit 2 Containment MC/CC Inspection Interval[4]</th></tr>
<tr><td>First Period [2,3]</td><td>Second Period [2,3]</td><td>Third Period [2,3]</td></tr>
<tr><td>5/11/2018 – 5/10/2021</td><td>5/11/2021 – 5/10/2024</td><td>5/11/2024 – 5/10/2028[1]</td></tr>
<tr><td>Outage 1 (B2R24)</td><td rowspan="2">Outage 3 (B2R26)</td><td>Outage 4 (B2R27)</td></tr>
<tr><td>Outage 2 (B2R25)</td><td>Outage 4 (B2R28)</td></tr>
</table>

Notes:

1. Unit 1 Fifth Interval end date for Class MC and CC components may be extended to no later than May 10, 2029 and may be reduced to no earlier than May 10, 2027. These provisions do not apply to inspection interval end dates for Class 1, 2, and 3 components.

2. Inspection periods for Subsection IWL (Class CC components) shall comply with the requirements of IWL-2410, based on a rolling 5-year frequency (+/- 1 year) from the completion date of the Structural Integrity Test (SIT). During the 5th Interval, IWL Examinations shall be performed only during the following periods:

a. Unit 1:

- IWL Period 1: August 20, 2020 through August 20, 2022 (i.e., includes Refueling Outage B1R24)

- IWL Period 2: August 20, 2025 through August 20, 2027 (i.e., includes Refueling Outage B1R26)

b. Unit 2:

- IWL Period 1: October 8, 2018 through October 8, 2020 (i.e., includes Refueling Outage B2R24)

- IWL Period 2: October 8, 2023 through October 8, 2025 (i.e., includes Refueling Outage B2R27)

3. The duration of the inspection periods has been adjusted, as permitted by IWA-2430(c)(3), to allow for Period 1 to have two (2) refueling outages, Period 2 to have a single (1) refueling outage, and Period 3 to have two (2) refueling outages.

4. The subsequent (i.e., 3rd) BSEP Inservice Inspection Interval Program for Class MC and CC components started on May 11, 2018, along with the start of the 5th Inservice Inspection Interval Program for Class 1, 2 and 3 components. To avoid confusion with regard to which inspection interval is being implemented, BSEP shall hereafter refer to the inspection interval starting on May 11, 2018 as the Fifth Inservice Inspection Interval for Class 1, 2, 3, MC and CC components:

3.2.1 IWE Program:

In Reference 5.1, the licensee stated that the applicable requirements of Subsection IWE for Class MC and Metal Liners of Class CC Components inservice inspection schedule is indicated per the Tables below:

Table 3.2.1-1 Unit 1 and Unit 2 Category E-A Examinations

Item Numbers	Parts Examined	Number of Items (per unit)	Number of Examinations Scheduled by Period (per unit)		
			1	2	3
E1.10 E1.11	Containment Vessel Pressure Retaining Boundary Accessible Surface Areas	203	203	203	203
E1.12	Wetted Surfaces of Submerged Areas	25	11	9	5
E1.20	BWR Vent System Accessible Surface Areas	19	0	0	19
E1.30	Moisture Barrier	1	1	1	1

Note: Portions of the surfaces (including bolted connections) of electrical penetrations are considered inaccessible for general visual examination in accordance with Category E-A, E1.11 because welded electrical junction boxes are attached just off of the containment wall, not allowing sufficient space to perform this visual examination. Surfaces are considered to be accessible for visual examination if the examination can be performed, either directly or remotely, by line of sight from available viewing angles from floors, platforms, walkways, ladders, or other permanent vantage points, as specified in IWE-2310(c).

Table 3.2.1-2 Unit 1 and Unit 2 Category E-C Requiring Augmented Examinations					
Item Numbers	Parts Examined	Number of Items (per unit)	Number of Examinations Scheduled by Period (per unit)		
			1	2	3
E4.10 & E4.11	Containment Surface Areas – Visual Surfaces	0	0	0	0
E4.12	Surface Area Grid - Minimum Wall Thickness Location	0	0	0	0

Note: At the start of the Fifth ISI Interval, there were no items identified requiring examination in accordance with Category E-C, Items E4.11 or E4.12. During the previous (Second) Containment ISI Interval, there were some Unit 2 items that required examination in accordance with Category E-C, but the requirements of IWE-2420(d) have been satisfied so these examinations are no longer required.

Table 3.2.1-3 E-G Examinations						
	Item Numbers	Parts Examined	Number of Items	Number of Examinations Scheduled by Period		
				1	2	3
Unit 1	E8.10	Bolted Connections	16	4	5	7
Unit 2	E8.10	Bolted Connections	16	5	11	0

Note: 100 percent of the examinations are required to be performed by the end of the Interval. Deferral to the end of the interval is permissible. Examinations may be performed at any time during the interval and may be performed with the connection assembled and bolting in place under tension, provided the connection is not disassembled during the interval. If the bolted connection is disassembled for any reason during the interval, the examination shall be performed with the connection disassembled.

3.2.2 IWL Program

'In Reference 5.1, the licensee stated that the third 10-year interval of concrete containment examinations (IWL) with Category L-A became effective on May 11, 2018 and will end on May 10, 2028. The inservice inspection schedule is indicated in the Table below:

Table 3.2.2 Unit 1 and Unit 2 Category L-A Examinations		
Item Numbers	Parts Examined	Areas required to be Examined During Each Period (per unit)
L1.10 L1.11	Concrete Surface All Accessible Surface Areas	100% (18 of 18 Areas)
L1.12	Suspect Areas	100% (if any)

Note: The IWL examination periods are as follows and do not align with those for Class 1, 2, 3 and MC Components:

a. Unit 1: IWL Period 1 – August 20, 2020 to August 20, 2022
IWL Period 2 – August 20, 2025 to August 20, 2027

b. Unit 2: IWL Period 1 – October 8, 2018 to October 8, 2020
IWL Period 2 – October 8, 2023 to October 8, 2025

3.2.3 Coating Program

In Reference 5.1, the licensee stated that the Nuclear Coatings Program provides a standardized method of selecting, procuring, applying, maintaining, and periodically assessing coatings so they can be used to minimize the adverse impacts of degraded (i.e., detached) coatings on SSCs, minimize material degradation of SSCs, facilitate decontamination of SSCs, and satisfy licensing and regulatory commitments.

The licensee also stated that service level I coatings are applied to exposed surfaces inside the reactor containment where coating failure (i.e., detachment) could adversely affect the operation of post-accident fluid systems and thereby, impair safe shutdown. Systems which could be impacted by detached coatings include the emergency core cooling system (ECCS) and the containment spray system. Therefore, coating systems used within the reactor containment are required to be qualified coating systems.

The licensee further stated that site engineering is responsible for verifying that the amount of unqualified coatings allowed in the primary containment does not exceed limits defined in calculations and the Updated Final Safety Analysis Report. A calculation is performed on the ECCS to ensure the amount of unqualified coatings in containment does not degrade the ECCS system. Areas where coatings are considered degraded are documented and tracked by a coatings exempt log to ensure that the quality of unqualified coatings in the torus do not exceed analyzed limits.

The licensee also stated that the total amount of protective coating debris or unqualified coatings inside the primary containment is limited to 6,667 cubic feet (ft^3) for each unit. The Maintenance Rule (a)(1) limit of 5.175 ft^3 has been set as an alert limit for each unit. This quantity was considered acceptable based on ECCS suction strainer design analysis. The estimated volumes of unqualified coatings inside the Unit 1 and Unit 2 BSEP primary containments are summarized in tables below.

Table 3.2.3-1 Unit 1 Primary Containment Unqualified Coatings following B1R22 Refueling Outage, March 2016	
Design Limit	6.667 ft^3
Maintenance Rule (a)(1) Limit	5.175 ft^3
Total volume following B121R1	3.120 ft^3
Percent of Maintenance Rule (a)(1) limit at start up	60.5%

Table 3.2.3-2 Unit 2 Primary Containment Unqualified Coatings following B2R23 Refueling Outage, March 2017	
Design Limit	6.667 ft^3
Maintenance Rule (a)(1) Limit	5.175 ft^3
Total volume following B121R1	3.647ft^3
Percent of Maintenance Rule (a)(1) limit at start up	70.5%

Based on the review of these tables, the NRC staff finds the licensee is properly implementing the ASME Section XI, Subsection IWE, IWL, and the coating programs, which provides reasonable assurance that the structural integrity of the concrete containment will be maintained if the ILRT frequency is extended to 15 years.

3.2.4 Containment In-Service Inspection Results Evaluation Summary

The NRC staff determined that the licensee's containment inspection programs support extension of the ILRT frequency as requested in the licensee's submittal of February 27, 2019. The NRC staff finds that there is reasonable assurance that the structural integrity of the BSEP Units 1 and 2 primary containment will continue to be monitored and maintained with the performance-based Type A test interval extended up to one test in 15 years, without undue risk to public health and safety because: (1) the licensee has adequately implemented its CISI Program to periodically examine, monitor, and manage the condition of its containment structure; and (2) the results of past containment structure visual inspections demonstrate acceptable performance of the containment and demonstrate that the structural integrity of the containment structure is adequate. Therefore, the NRC staff concludes that the licensee's containment inspection programs support the proposed LAR to change TS 5.5.12 to extend the integrated leakage rate test frequency to 15 years for Type A tests on a permanent basis, and the licensee's adoption of NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, as the documents used by BSEP Units1 and 2 to implement the performance based leakage testing program in accordance with Option B of 10 CFR Part 50, Appendix J.

3.3 Supporting Plant-Specific Risk Impact Assessment Evaluation

Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of NEI 94-01, Revision 3-A (ADAMS Accession No. ML122210254), "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," discusses how plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," of NEI 94-01 states that "the assessment should be performed using the approach and methodology described in EPRI Technical Report (TR) 1018243[2], "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval."

In the Final Safety Evaluation (SE), dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC staff found the methodology in NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, acceptable for referencing by licensees proposing to amend their TS to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied. These conditions, set forth in Section 4.2 of the SE for EPRI TR-1009325, Revision 2, stipulate that:

1. The licensee submit documentation indicating that the technical adequacy of their Probabilistic Risk Assessment (PRA) is consistent with the requirements of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," relevant to the ILRT extension application.

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[2] EPRI TR-1018243, is also identified as EPRI TR-1009325, Revision 2-A. This report is publicly available and can be found at www.epri.com by typing "1018243" in the search field box.

[Additional application specific guidance on the technical adequacy of a PRA used to extend ILRT intervals is provided in the SE for EPRI TR-1009325, Revision 2.]

2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small and consistent with the clarification provided in Section 3.2.4.6[3] of the SE for EPRI TR-1009325, Revision 2.

3. The methodology in EPRI TR-1009325, Revision 2, is acceptable provided the average leak rate for the pre-existing containment large leak accident case (i.e., accident case 3b) used by licensees is assigned a value of 100 L_a instead of 35 L_a.

4. A [license amendment request] LAR is required in instances where containment over-pressure is relied upon for [emergency core cooling system] ECCS performance. According to the clarification provided in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, plants that rely on containment over-pressure (or containment accident pressure) net positive suction head (NPSH) for ECCS injection must also consider core damage frequency (CDF) in the ILRT evaluation.

In the LAR, the licensee provided a risk assessment for permanently extending the currently allowed containment Type A ILRT interval from 10 years to 15 years in Section 3.5.1 of Enclosure 1 and Enclosure 6 to the LAR submitted February 27, 2019.

In Section 3.5.1 of Enclosure 1 and Enclosure 6 to the LAR, the licensee stated that the plant-specific risk assessment follows the guidance in NEI 94-01, Revision 3-A (ADAMS Accession No. ML12221A202); the methodology described in EPRI TR-1018243 (also identified as EPRI TR-1009325, Revision 2-A); and, the NRC regulatory guidance outlined in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Additionally, the licensee applied the methodology from Calvert Cliffs Nuclear Power Plant (ADAMS Accession No. ML020920100) to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during extended test interval.

The licensee addressed each of the four conditions for the use of EPRI TR-1009325, Revision 2, which are listed in Section 4.2 of the NRC SE. A summary of how each condition is met is provided in Sections 3.3.1 thru 3.3.4 below.

3.3.1 PRA Quality – Condition 1

The first condition in Section 4.2 of the SE for EPRI TR-1009325, Revision 2 stipulates that the licensee submit documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

Consistent with the information provided in Regulatory Issue Summary (RIS) 2007-06 (ADAMS Accession No. ML070650428), "Regulatory Guide 1.200 Implementation," the NRC staff will use Revision 2 of RG 1.200 (ADAMS Accession No. ML090410014) to assess technical adequacy of the PRA used to support risk-informed applications received after March 2010. In

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[3] Section 4.2 of the SER for EPRI TR-1009325, Revision 2, indicates that the clarification regarding small increases in risk is provided in Section 3.2.4.5; however, the clarification is actually provided in Section 3.2.4.6.

Section 3.2.4.1 of the SE for NEI 94-01, Revision 2 and EPRI TR-1009325, Revision 2, the NRC staff states that Capability Category (CC) I of the American Society of Mechanical Engineers (ASME) PRA standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

As discussed in Section 3.5.2 of Enclosure 1 and Appendix A of Attachment 6 of the LAR, the BSEP risk assessment performed to support the ILRT application uses the current BSEP Level 1 and Level 2 internal events PRA (IEPRA) model of record, which the licensee completed in April 2010. The IEPRA also includes internal flooding hazards.

The licensee stated that the BSEP IEPRA underwent a full-scope peer review in 2010 against the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard, RA-Sa-2009, and the clarifications of RG 1.200, Revision 2 consistent with NRC RIS 2007-06 (ADAMS Accession No. ML070650428). The purpose of this review was to establish the acceptability of the PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used. A focused scope peer review of the Internal Flood model was conducted in December 2016, which covered 28 Supporting Requirements (SRs).

The ASME PRA Standard has 325 individual SRs; 322 SRs are applicable to the BSEP PRA. Three of the ASME/ANS PRA Standard Supporting Requirements are not applicable to BSEP (e.g., PWR-related, linked event tree methodology-related). Of the 322 ASME/ANS PRA Standard SRs applicable to BSEP, approximately 88 percent are supportive of CC II or greater. Of the 79 unique Facts and Observations (F&Os) generated by the Peer Review Team, 44 were considered peer review Findings and 35 were Suggestions.

An F&O closure technical review was conducted in August 2017 and most, but not all, open F&Os were reviewed for closure. The review applied the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," (ADAMS Accession No. ML17086A431), as accepted by the NRC staff in a letter dated May 3, 2017, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute, Appendix X to Guidance 05-04, 7-12, and 12-13, Close Out of Facts and Observations (F&Os)," (ADAMS Accession No. ML17079A427). Following the F&O closure, 15 Internal Event and Internal Flooding F&Os remain open; these F&Os are dispositioned in Section A.1.1 of Attachment 6 of the LAR for their impact on the ILRT extension. The licensee determined all the Finding Level F&Os would not significantly affect the ILRT extension analysis. The NRC staff reviewed these F&Os and the associated dispositions and determined that they will not significantly impact on the ILRT application.

In Section 3.2.4.2 of the SE for NEI 94-01, Revision 2 and EPRI TR-1009325, Revision 2, the NRC staff states that:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, “External Events,” states that: “Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals.” This section also states that: “If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [(i.e., EPRI Report No. 1009325,

> Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed." This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

Therefore, the NRC staff's review of the contribution of external events for this application is framed by the context that an order of magnitude estimate for the corresponding risk contribution is sufficient. For the purposes of this analysis, the licensee's external event models are the Fire PRA and High Winds PRA models. Diverse and Flexible Coping Strategies (FLEX) were not modeled in the External Events models; and therefore, was not credited in the External Events loss of containment accident pressure analysis. Crediting FLEX would reduce the increase due to a loss of containment accident pressure for High Winds events, since FLEX is a separate form of late injection that would likely be available even if the intermittent long-term use of the low pressure ECCS systems was failed due to a loss of containment accident pressure.

The BSEP Fire Probabilistic Risk Assessment (FPRA) Peer Review was performed December 2011 using the NEI 07-12 process, the ASME PRA Standard, and RG 1.200, Revision 2. The purpose of this review was to establish the acceptability of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The 2011 BSEP FPRA Peer Review was a full-scope review of all the technical elements of the BSEP at-power 2011 Fire PRA against all technical elements in Section 4 of the ASME/ANS Combined PRA Standard, including the referenced internal events SRs in Section 2 of the ASME/ANS Combined PRA Standard.

The Fire PRA Section of the ASME PRA Standard has 182 individual SRs, and references 237 individual SRs in the internal events PRA section of the Standard; the BSEP Peer Review included all SRs and all applicable reference SRs (i.e., see Table A.2-1 of Attachment 6 of the LAR). For the assessment of the reviewed ASME PRA Standard SRs, 105 unique F&Os were generated by the Peer Review team, 53 were peer review Findings, 50 were Suggestions, and one was an "Unreviewed Analytical Method."

An F&O Finding closure technical review was conducted in July 2017 to review all open Fire PRA F&Os. Following the F&O closure, six F&Os remain open; these F&Os are dispositioned in Section A.2.1 of Attachment 6 of the LAR for their impact on the ILRT extension. The licensee determined all the Finding Level F&Os would not significantly affect the ILRT extension analysis. The NRC staff reviewed these F&Os and the associated dispositions and determined that they have no impact on the ILRT application.

Based on review of the above information, the NRC staff finds the licensee has addressed the relevant findings and gaps from the peer reviews and that they have no impact on the results of this application. Therefore, the NRC staff concludes that the internal events PRA model used by the licensee is of sufficient quality to support the evaluation of changes to ILRT frequencies. Accordingly, the first condition is met.

3.3.2 Estimated Risk Increase – Condition 2

The second condition stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points. Lastly, for plants that rely on containment over-pressure for net positive suction head (NPSH) for ECCS injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. This RG defines very small changes in risk as resulting in increases of CDF and LERF of less than 1.0E-6/year and 1.0E-07/year respectively. Original plant NPSH calculations took no credit for containment (i.e., suppression chamber) pressure. As part of the 120 percent power uprate, the commitment was revised. Specifically, a 5 pounds per square inch gauge (psig) credit for containment overpressure was established as acceptable for evaluating low pressure ECCS pump NPSH. Thus, the associated risk metrics include LERF, population dose, CCFP, delta CDF and LERF.

The licensee reported the results of the plant-specific risk assessment in Section 3.5.3 of Enclosure 1 and Section 6.0 of Appendix 6 to the LAR. Details of the risk assessment are provided in Appendix 6. The reported risk impacts are based on a change in the Type A containment ILRT frequency from three tests in 10 years (the test frequency under 10 CFR Part 50, Appendix J, Option A) to one test in 15 years and account for the risk from undetected containment leaks due to steel liner corrosion. The following conclusions can be drawn from the licensee’s analysis associated with extending the Type A ILRT frequency:

1. RG 1.174 defines very small changes in risk as resulting in increases of CDF less than 1.0E-6/year. Since BSEP relies on containment accident pressure for ECCS NPSH during certain design basis accidents, extending the ILRT interval may impact CDF. Therefore, the BSEP PRA model was used to estimate the potential change in CDF if containment accident pressure was unavailable due to a pre-existing containment leak. The containment accident pressure analysis discussed in Section 5.2.7 of Attachment 6 of the LAR estimates the potential increase in the overall CDF to be 1.61E-08/year for Unit 1 and 1.54E-08/year for Unit 2. Thus, the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small using the acceptance guidelines of RG 1.174.

   RG 1.174 defines very small changes in risk as resulting increases in LERF less than 1.0E-07/year. The increase in LERF resulting from a change in the Type A ILRT test interval from 3-in-10 years to 1-in-15 years is estimated as 6.03E-08/year for Unit 1 and 5.67E-08/year for Unit 2 using the EPRI guidance; this value increases negligibly if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included. As such, the estimated change in LERF is determined to be very small using the acceptance guidelines of RG 1.174. When external event risk is included, the increase in LERF resulting from a change in the Type A ILRT test interval from 3-in-10 years to 1-in-15 years is estimated as 4.05E-07/year for Unit 1 and 4.33E-07/year for Unit 2 using the EPRI guidance, and total LERF is 5.55E-06/year for Unit 1 and 5.53E-06/year for Unit 2. As such, the estimated change in LERF is determined to be small using the acceptance guidelines of RG 1.174. The risk change resulting from a change in the Type A ILRT test interval from 3-in-10 years to 1-in-15 years bounds the 1-in-10 years to 1-in-15 years risk change.

2. The effect resulting from changing the Type A test frequency to 1-per-15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 4.98E-03 person-rem/year for Unit 1 and 4.67E-03 person-rem/year for Unit 2. NEI 94-01 states that a small total population dose is defined as an increase of ≤ 1.0 person-rem/year, or ≤ 1% of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The reported increase in total population dose is below the acceptance criteria provided in EPRI TR-1009325, Revision 2-A, and defined in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2. Thus, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

3. The increase in the CCFP due to the change in test frequency from three in 10 years to once in 15 years is 1.20 percent for Unit 1 and 1.213 percent for Unit 2. NEI 94-01 states that increases in CCFP of ≤ 1.5 is small. These values are below the acceptance guidelines in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2 and supportive of the proposed change.

Based on the risk assessment results, the NRC staff concludes that the increase in LERF is small and consistent with the acceptance guidelines of RG 1.174, and the increase in the total population dose and the magnitude of the change in the CCFP for the proposed change are small. The defense-in-depth philosophy is maintained as the independence of barriers will not be degraded because of the requested change, and the use of the quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, the second condition is met.

3.3.3 Large Pre-Existing Containment Leak Rate Case – Condition 3

The third condition stipulates that to make the methodology in EPRI TR-1009325, Revision 2 acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a. As noted by the licensee in Section 3.5.1 of Enclosure 1 to the LAR, the methodology in EPRI TR-1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the BSEP plant-specific risk assessment. Accordingly, the third condition is met.

3.3.4 ECCS Performance Reliance on Containment Overpressure – Condition 4

The fourth condition stipulates that in instances where containment over-pressure is relied upon for ECCS performance, a LAR is required to be submitted. In Section 3.3 of Enclosure 1 of the LAR, the licensee stated in part that:

> As part of the 120 percent power uprate, the commitment was revised. Specifically, a 5-pound per square inch gauge (psig) credit for containment overpressure was established as acceptable for evaluating low pressure Emergency Core Cooling System (ECCS) pump NPSH. The 120 percent power uprate was approved by the NRC on May 31, 2002 [(ADAMS Accession No. ML021430551)].

According to the clarification provided in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2 plants that rely on containment over-pressure

NPSH for ECCS injection must also consider CDF in the ILRT evaluation. Since containment pressure is credited in support of ECCS performance to mitigate design basis accidents at BSEP, the licensee reported on a detailed analysis, described in Attachment 6, Section 5.2.7, to quantify the potential effect on CDF. The acceptance guidelines in RG 1.174 were used to assess the acceptability of this permanent extension of the Type A Test interval beyond that established by Option B of Appendix J. RG 1.174 defines very small in the risk-acceptance guidelines as increases in CDF less than 10^{-6} per reactor year. The results of these analyses are discussed in Attachment 6, Section 6 of the LAR. The increase in CDF reported in the LAR was determined to meet the guidelines in RG 1.174. Accordingly, the fourth condition is met.

3.3.5 Supporting Plant-Specific Risk Impact Assessment Evaluation Summary

The NRC staff reviewed the response to the four risk related conditions stipulated in the acceptance of the methodology in NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2. The NRC staff found these conditions are satisfied.

3.4 Technical Review Summary

The NRC staff reviewed these aspects of the application to ensure the applicable limitations and conditions of NEI 94-01, Revisions 2-A and 3-A have been adequately addressed. The evaluation supports approval of the requested changes, because the required containment leak-tight integrity will be maintained.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the appropriate official for the State of North Carolina was notified of the NRC's proposed issuance of the amendments on December 11, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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, and there has been no public comment on such finding (84 FR 19967, dated May 7, 2019). 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.

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Date: February 6, 2020

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 298 AND 326 TO MODIFY TECHNICAL SPECIFICATIONS TO EXTEND CONTAINMENT LEAKAGE TEST INTERVALS (EPID L-2019-LLA-0031) DATED FEBRUARY 6, 2020

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