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

October 11, 2016

Mr. Richard M. Glover
Site Vice President
H. B. Robinson Steam Electric Plant
Duke Energy Progress, LLC
3581 West Entrance Road, RNPA01
Hartsville, SC 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENT TO EXTEND CONTAINMENT LEAKAGE RATE TEST FREQUENCIES (CAC NO. MF7102)

Dear Mr. Glover:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 247 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2). This amendment changes the HBRSEP2 Technical Specifications (TSs) in response to your application dated November 19, 2015, as supplemented by letter dated August 18, 2016.

The amendment revises HBRSEP2 TS 5.5.16, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," and Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option A, with a reference to Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as the implementation documents used by HBRSEP2 for the performance-based leakage testing program in accordance with 10 CFR Part 50, Appendix J, Option B. The amendment also includes an administrative change to TS 5.5.16 that deletes the information regarding the performance of the next Type A test no later than April 9, 2007, as this test has already occurred.

The amendment also revises Surveillance Requirements 3.6.1.1 and 3.6.2.1, by replacing references to 10 CFR Part 50, Appendix J, Option A, with references to the Containment Leakage Rate Testing Program.

The TS changes allow the extension of the Type A test interval from 10 years to 15 years and the extension of the Type B and Type C test intervals for selected components to 120 months and 75 months, respectively. The TS changes also adopt a grace interval of 9 months for Type A, Type B, and Type C tests in accordance with NEI 94-01, Revision 3-A.

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,

Dennis J Galvin

Dennis J. Galvin, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 247 to DPR-23
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 247
Renewed License No. DPR-23

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

   A. The application for amendment by Duke Energy Progress, LLC (the licensee) (previously Duke Energy Progress, Inc.), dated November 19, 2015, as supplemented by letter dated August 18, 2016, 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.

Enclosure 1

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment. Paragraph 3.B. of Renewed Facility Operating License No. DPR-23 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 247, are hereby incorporated in the license.

      The licensee 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeanne A. Dion

Jeanne A. Dion, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-23
and Technical Specifications

Date of Issuance: October 11, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 247

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace page 3 of Renewed Facility Operating License No. DPR-23 with the attached page 3.

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

Remove Pages	Insert Pages
3.6-2	3.6-2
3.6-5	3.6-5
3.6-6	3.6-6
5.0-22	5.0-22

D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by 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, Section 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 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:

A. Maximum Power Level

The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of 2339 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 247 are hereby incorporated in the license.

The licensee shall operate the facility in accordance with the Technical Specifications.

(1) For Surveillance Requirements (SRs) that are new in Amendment 176 to Final Operating License DPR-23, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 176. For SRs that existed prior to Amendment 176, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 176.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3 ----------NOTE-------------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ------------------------------- Verify an OPERABLE door is locked closed.	Once per 31 days
C. Containment air lock inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	<u>AND</u>	
	C.2 Verify a door is closed in the air lock.	1 hour
	<u>AND</u>	
	C.3 Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.1	-----NOTES----- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program.
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	24 months

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program

A program shall establish the leakage rate testing of the 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 Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

The peak containment pressure, P_a, is specified as the containment design pressure of 42 psig. The containment design pressure is 42 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of the containment air weight per day.

Leakage rate acceptance criteria are:

1. 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 $\leq$ 0.60 L_a for the Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:
   a. Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq P_a$.
   b. For each door, leakage rate is $\leq$ 0.01 L_a when pressurized to $\geq$ 42 psig.

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

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.17 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be implemented to ensure that, with an OPERABLE Control Room Emergency Filtration System, CRE occupants can control the nuclear power unit safely following a radiological event, hazardous chemical release, or a smoke challenge. 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. 247 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DUKE ENERGY PROGRESS, LLC

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By application dated November 19, 2015 (Reference 1), as supplemented by letter dated August 18, 2016 (Reference 2), Duke Energy Progress, LLC, the licensee (previously Duke Energy Progress, Inc.), submitted a license amendment request (LAR) for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2). The LAR proposed to revise HBRSEP2 Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 3), and Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option A, "Prescriptive Requirements," with a reference to Nuclear Energy Institute (NEI) topical report (TR) NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012 (Reference 4), and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008 (Reference 5), as the implementation documents used by HBRSEP2 for the performance-based leakage testing program in accordance with 10 CFR Part 50, Appendix J, Option B, "Performance-Based Requirements." The LAR also proposed an administrative change to TS 5.5.16 that would delete the information regarding the performance of the next Type A (integrated leak rate) test no later than April 9, 2007, as this test has already occurred.

The LAR also proposed revising Surveillance Requirements (SRs) 3.6.1.1 and 3.6.2.1, of TS 3.6.1, "Containment," and TS 3.6.2, "Containment Air Lock," respectively, by replacing references to 10 CFR Part 50, Appendix J, Option A, with references to the Containment Leakage Rate Testing Program.

In accordance with the guidance in NEI 94-01, Revision 3-A, and the limitations and conditions for NEI 94-01, Revision 2-A, the proposed changes would permit the performance-based primary containment integrated leak rate testing (ILRT), also known as a Type A test, maximum interval to be extended from no longer than 10 years to no longer than 15 years, provided acceptable performance history and other requirements stated in the topical report are maintained. In accordance with NEI 94-01, Revision 3-A, the proposed change would also permit the containment isolation valve local leakage rate tests (LLRTs), also known as a Type C test, maximum interval to be extended to 75 months and the Type B (penetration barriers other than valves) LLRTs maximum interval to be extended to 120 months.

The licensee justified the proposed TS changes by providing historical plant-specific Containment Leakage Testing Program results and Containment Inservice Inspection (CISI) Program results and a supporting plant-specific risk assessment, consistent with the guidance in NEI 94-01, Revision 3-A, and the conditions and limitations contained in NEI 94-01, Revision 2-A.

The supplemental letter dated August 18, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 15, 2016 (81 FR 13841).

2.0 REGULATORY EVALUATION

Section 50.54(o) of 10 CFR requires that primary reactor containments for water-cooled power reactors be subject to the requirements set forth in Appendix J to 10 CFR Part 50. The regulations in 10 CFR Part 50, Appendix J, include 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 the appendix.

The testing requirements in 10 CFR Part 50, Appendix J, ensure that (a) leakage through containments or systems and components penetrating containments does not exceed allowable leakage rates specified in the TSs and (b) integrity of the containment structure is maintained during its service life. HBRSEP2 adopted 10 CFR Part 50, Appendix J, Option B, for the Type A test by License Amendment No. 169, dated May 28, 1996 (Reference 6), but retained Option A for the Type B and Type C tests.

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's 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 containment isolation valve 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 Type 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 allowable leakage rate (L_a) with margin, as specified in the TSs. Option B also requires that a general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be conducted prior to each Type A test and at a periodic interval between tests based on the performance of the containment system.

Section V.B.3 of 10 CFR Part 50, Appendix J, Option B, requires that the RG 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 and endorsed in an RG.

The implementation document that is currently referenced in HBRSEP2 TS 5.5.16, is RG 1.163, which endorsed NEI 94-01, Revision 0, dated July 26, 1995 (Reference 7). NEI 94-01, Revision 0, provides methods acceptable to the NRC staff for complying with the provisions of Option B of 10 CFR Part 50, Appendix J, subject to four regulatory positions delineated in Section C of RG 1.163 and also includes provisions that allow the performance-based Type A test interval to be extended up to 10 years, based upon two consecutive successful tests.

NEI 94-01, Revision 2 and Revision 3 (Reference 8 and Reference 9), have been reviewed by the NRC and approved for use. They incorporate the regulatory positions stated in RG 1.163 and include provisions for extending Type A test intervals to up to 15 years. The final safety evaluation (SE) for Revision 2, issued by letter dated June 25, 2008 (Reference 10), documents the NRC's evaluation and acceptance of Revision 2, subject to six specific limitations and conditions listed in Section 4.1 of the SE. The final SE for Revision 3, issued by letter dated June 8, 2012 (Reference 11), includes two specific limitations and conditions listed in Section 4.0 of the SE for the Type C test. Revision 2-A and Revision 3-A of NEI 94-01 include the corresponding NRC staff SEs.

The regulations at 10 CFR 50.55a "Codes and standards," contain the CISI Program requirements that, in conjunction with the requirements of Appendix J of 10 CFR Part 50, ensure the continued leak-tightness and structural integrity of the containment during its service life.

The regulations at 10 CFR 50.65(a)(1), "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," state, in part, that the licensee:

> ... shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience.

The regulations at 10 CFR 50.36, "Technical specifications," state that the TSs include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) SRs; (4) design features; and (5) administrative controls.

NUREG-1431, Revision 4, "Standard Technical Specifications - Westinghouse Plants," dated April 2012 (Reference 12), incorporated the Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-52, Revision 3, "Implement 10 CFR 50, Appendix J, Option B" (Reference 13), which provided guidance for specific changes to TSs for implementation of 10 CFR Part 50, Appendix J, Option B.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 14) describes one acceptable approach for determining whether the technical adequacy of the probabilistic risk assessment (PRA), in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. It is also intended to reflect and endorse guidance provided by standards-setting and nuclear industry organizations.

Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation" (Reference 15), clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200 (Reference 16) to assess technical adequacy of the PRA used to support risk-informed applications. Revision 2 of RG 1.200 (Reference 14) will be used for all risk-informed applications received after March 2010.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 17) defines an acceptable approach for use in analyzing and evaluating proposed licensing basis changes. This approach supports the NRC staff's desire to base its decisions on the results of traditional engineering evaluations, supported by insights (derived through the use of PRA methods) on the risk significance of the proposed changes. The decision-making process leading to the proposed change is expected to follow an integrated approach (considering traditional engineering and risk information) and may build upon qualitative factors as well as quantitative analyses and information.

The LAR included TS Bases sections corresponding to the proposed changes. As the TS Bases are a licensee-controlled document, this material was reviewed only for general consistency with the proposed TS changes.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Changes

HBRSEP2 TS 5.5.16 currently states:

> This program provides controls for implementation of the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions for Type A testing. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception:
>
> a. NEI 94-01 – 1995, Section 9.3.2: The first Type A test performed after the April 9, 1992, Type A test shall be performed no later than April 9, 2007.
>
> Type B and Type C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.
>
> The peak containment pressure, P_a, is specified as the containment design pressure of 42 psig [pounds per square inch gauge], which exceeds the peak calculated containment internal pressure for the design basis loss of coolant accident.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of the containment air weight per day.

Leakage rate acceptance criteria are:

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

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

With the proposed change, TS 5.5.16 would state:

A program shall establish the leakage rate testing of the 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 Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

The peak containment pressure, P_a, is specified as the containment design pressure of 42 psig. The containment design pressure is 42 psig.
The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of the containment air weight per day.

Leakage rate acceptance criteria are:

1. 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 $\leq$ 0.60 L_a for the Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:
   a. Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a.
   b. For each door, leakage rate is $\leq$ 0.01 L_a when pressurized to $\geq$ 42 psig.

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

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

With the proposed change, HBRSEP2 will implement NEI 94-01, Revision 3-A, and the limitations and conditions of Section 4.1 of the NEI 94-01, Revision 2-A, SE. NEI 94-01, Revision 3-A, provides that extension of the Type A test 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 (1) implementation of robust Type B and Type C testing of the penetration barriers, where most containment leakage has historically been shown to occur and is expected to continue to be the pathway for a majority of potential primary containment leakage, and (2) implementation of a robust containment visual inspection program where deterioration of the primary containment boundary away from penetrations can be detected and remediated before any actual significant leakage may occur.

NEI 94-01, Revision 3-A, also provides that a maximum interval of 120 months could be allowed for Type B tests, and a maximum interval of 75 months could be allowed for Type C tests, with a limited provision for extension or grace period up to 9 months allowed for these LLRTs.

The existing TS 5.5.16 exception to NEI 94-01 regarding the performance of the next Type A test by April 9, 2007, is no longer needed and may be removed, as that date has passed with the test having been completed.

The reference to RG 1.163 is to be replaced with a reference to NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A. This is allowed by the provision in 10 CFR Part 50, Appendix J, Option B, Section V.B.3, regarding the TSs referencing the NRC staff-approved guidance document for program implementation.

HBRSEP2 TS SR 3.6.1.1 currently states:

> Perform required Type B and C leakage rate testing except for containment air lock testing, in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.
>
> The leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions, the leakage rate acceptance criterion is $< 0.6\ L_a$ for the Type B and Type C tests.

With the proposed change, TS SR 3.6.1.1 would state:

> Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.

HBRSEP2 TS SR 3.6.1.3 currently states:

> Perform required visual examinations and Type A leakage rate testing, in accordance with the Containment Leakage Rate Testing Program.

With the proposed change, TS SR 3.6.1.3 would be deleted, as the requirement would be incorporated into TS SR 3.6.1.1.

HBRSEP2 TS SR 3.6.2.1 currently states:

---------------------------NOTES---------------------------------

1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

2. Results shall be evaluated against acceptance criteria of SR 3.6.1.1, in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.

Perform required air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.

With the proposed change, TS SR 3.6.2.1 would state:

---------------------------NOTES---------------------------------

1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.

Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

The proposed change would also remove the NOTE, "SR 3.0.2 is not applicable," from the Frequency column of TS SR 3.6.1.1 and SR 3.6.2.1.

The NRC staff finds that the LAR proposed changes to TS SR 3.6.1.1, SR 3.6.1.3, and SR 3.6.2.1 are consistent with the implementation of 10 CFR Part 50, Appendix J, Option B, as reflected in the Standard Technical Specifications, NUREG-1431.

3.2 Deterministic Considerations – Structural and Leak-Tight Integrity of the Containment

In accordance with the LAR, the licensee proposed to extend the current performance-based Type A test interval from 10 years to 15 years by adopting NEI 94-01, Revision 3-A, and the conditions and limitations contained in NEI 94-01, Revision 2-A, as the implementation document in TS 5.5.16. This change would require HBRSEP2 to conduct the next Type A test by spring 2022,[1] in lieu of the current scheduled date of spring 2017. The licensee justified the proposed change by demonstrating adequate performance of the HBRSEP2 containment based on the plant-specific Containment Leakage Testing Program and CISI (IWE/IWL) Program results, and as supported by a plant-specific risk assessment, consistent with the guidance in NEI 94-01, Revision 3-A.

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[1] Due to the timing of planned Robinson outages, the LAR states that the next Type A test would be performed at subsequent refueling outage no later than spring 2021.

3.2.1 Description of the HBRSEP2 Reactor Containment Structure

HBRSEP2 is a Westinghouse design 3-loop pressurized water reactor within a large, dry ambient design containment. The HBRSEP2 reactor containment structure is a post-tensioned steel lined reinforced concrete shell in the form of a vertical right cylinder with a hemispherical dome and a flat base supported by piles with an inside diameter of 130 feet (ft) and a steel liner, which varies in thickness from 0.25 inches (in.) to 0.50 in. The structure sidewalls measure 126 ft. from the liner on the base to the springline of the dome with an inside diameter of 130 ft. The containment free volume is 1,950,000 ft^3. The inside radius of the dome is equal to the inside radius of the cylinder. The prestressing system chosen for post-tensioning the containment in the vertical direction consists of high strength steel bars closely grouped into tendons consisting of six bars per tendon placed within heavy wall 6 in. galvanized steel pipe sheaths. The LAR stated that the tendons were initially tensioned in April 1970 and then re-tensioned in May 1970. The sidewalls of the cylinder and dome are 3.5 ft. thick, the dome is 2.5 ft. thick, and the walls are reinforced circumferentially and prestressed vertically. The dome of the containment is reinforced concrete. The containment is supported on a 10 ft. thick structural concrete base mat. The base consists of a 10 ft. thick structural concrete slab (reinforced), and the base liner is installed on top of the structural slab and covered with 2 feet of concrete. In the vertical walls and dome, the liner is anchored to the concrete shell by means of "KSM" shaped anchor studs fusion welded to the liner plate forming an integral part of the entire composite structure under all loadings.

The containment structure is designed to contain radioactive material following a loss-of-coolant accident (LOCA), as described in Updated Final Safety Analysis Report (UFSAR) Section 6.2.1, and it also serves as a biological shield. TS 5.5.16 identifies the primary containment allowable leakage rate (L_a) as 0.1 weight percent of the contained free volume per day at the calculated maximum design-basis accident (DBA)-LOCA pressure (P_a) of 42 psig. The steel liner and its welded seam joints are covered by carbon steel channels with pressurizing connections used to determine the leak-tightness of the liner seam welds. The containment building and steel liner are designed to withstand fatigue, accident, and operational loading conditions, and load combinations as discussed in UFSAR Section 3.8.1.4. In accordance with UFSAR Section 3.8.1.6.1.5 and the LAR, the coated surfaces of the liner plate are protected by insulation and sheathing from elevation 228 ft up to elevation 367 ft 10 in. (i.e., the cylindrical portion of the liner). The containment liner insulation consists of 1.25 in. thick, 4 pound (lb)/ft^3 density, crosslinked PVC and/or Polyimide foam with an outer covering of 0.019 in. thick stainless steel.

3.2.2 HBRSEP2 Type A Test Performance History

In Section 3.2.4 of the LAR, the licensee stated that previous Type A tests confirmed that the reactor containment structure's leakage is well under acceptance limits and represents minimal risk to increased leakage, and that the CISI (IWE/IWL) Program and maintenance rule monitoring provide confidence in containment integrity. In LAR Table 3.2.4-1, the licensee presented the historical results of the Type A ILRT tests as summarized below.

Date	Leakage (Primary Containment Weight % per Day)
April 1992	0.0602
May 2007	0.0244

The licensee summarized the results of the last seven operational Type A tests performed at HBRSEP2 between May 1974 and May 2007. The initial test in 1974 was performed at full pressure, but until the 1992 test, the intervening four tests had been performed at ½ design (and DBA-LOCA calculated peak) pressure as allowed by 10 CFR Part 50, Appendix J, at the time (and continued in Option A). Direct comparison or trending with these ILRT results is difficult. However, they all appear to show margin to L_a even with very conservative extrapolation to a corresponding leakage at full pressure. Option B does not allow reduced pressure Type A tests. The licensee noted that the test results demonstrated considerable margin with the TS 5.5.16 acceptance limit of 0.75 L_a (L_a equals 0.1 percent by weight of the containment air per day at the peak accident pressure) demonstrating a low leakage containment.

The NEI 94-01, Revision 3-A, guidance for allowing the extended test interval is that the past two tests meet the performance criterion, showing a leakage of L_a or less. The test results further demonstrate that since the as-found (AF) results were less than 1.0 L_a, a test frequency of at least once per 15 years would be supported. In particular, the 1992 and 2007 ILRTs showed leakage to be 60 percent or less of the performance criterion (0.1 weight percent per day), so the extended interval would be allowed. The licensee stated in the LAR that based on previous ILRT tests conducted at HBRSEP2, it may be concluded that the permanent extension of the containment ILRT interval from 10 years to 15 years represents minimal risk to increased leakage testing performed in accordance with Option B of 10 CFR Part 50, Appendix J, and the overlapping inspection activities performed as part of the CISI (IWE/IWL) and primary containment coatings condition assessment programs.

The NRC staff concludes that since the seven Type A ILRTs performed to date were less than the design-basis leak rate, the licensee has demonstrated acceptable performance history pursuant to NEI 94-01, Revision 3-A. The results of the Type A ILRTs provide reasonable assurance that the containment overall leakage will be maintained below the design-basis leak rate, consistent with the requirements in TS 5.5.16.

3.2.3 Historical Type B and Type C Test (LLRT) Results

In LAR Table 3.3-1, the licensee presented the historical results of the Type B and Type C tests combined as-left (AL) maximum pathway leakage totals as summarized below:

Date	As-Left Maximum Pathway Leakage Rate (standard cubic centimeters per minute, sccm)	% of TS 5.5.16 Combined Type B and C Total Criterion (0.6 L_a which equates to 91,491 sccm)
Fall 2005	11985	13.1
Spring 2007	18392	20.1
Fall 2008	22972	25.1
Summer 2010	16983	18.6
Spring 2012	10436	11.4
Fall 2013	9844	10.8
Spring 2015	7122	7.9

The TS 5.5.16 acceptance criterion for combined Type B and Type C test total is 0.6 L_a, which the LAR indicated as being 91,491 sccm. These totals were calculated using the AL maximum pathway values. Option B has the AF minimum pathway values totaled and evaluated with the performance criterion. This was not explicitly required under Option A, but all LLRTs were performed each refueling outage under Option A, as there were no extended interval tests. The LAR indicated that a review of the Type B and Type C test results from 2007 through 2015 was performed that showed an exceptional amount of margin between the actual AF and AL combined Type B and Type C test totals. The AL maximum pathway totals shown are typical for the industry, and there is no apparent adverse trend. This suggests that the performance criteria are unlikely to be exceeded by use of extended LLRTs.

3.2.4 Containment Inservice Inspection Program (American Society of Mechanical Engineers (ASME) Code, Section XI, Subsections IWE and IWL)

In Section 3.2.2 of the LAR, the licensee described its ASME Code, Section XI, Subsection IWE, Inservice Inspection Program, and provided a schedule of examinations. Section 3.4.1 provided the results of recent examinations as a result of the licensee's review of NRC Information Notice (IN) 2010-12, "Containment Liner Corrosion" (Reference 18) and IN 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner" (Reference 19). The inspections were conducted primarily during the first and second 10-year intervals from 1998 - 2015. Past inspections of the liner have noted minor issues such as general surface corrosion, degraded protective coatings, and minor corrosion of the containment liner. The licensee noted that these items have shown no significant change over time and do not indicate any structural degradation that would adversely impact the ability of the containment to perform its design function.

In its review of IN 2014-07, the licensee stated that the HBRSEP2 system is not the same as that cited in the IN since it uses manifolds 3.5 ft above the floor, instead of a capped floor access box. The manifolds are vented to containment atmosphere and are sealed passages not conducive to flow to transport moisture. However, the licensee noted in its general review of the IN that a gap existed in the IWE program in that the piping/tubing runs that go through the concrete had not been previously included in the IWE inspection program to verify that there is no corrosion or a breach near the floor level, which would allow moisture to enter in a manner like the plants described in the IN. As a result, the licensee stated that the IWE program was revised to perform visual inspections of accessible tubing in the Penetration Pressurization System (PPS) from manifold to floor of the containment vessel.

The NRC staff reviewed the information provided in the LAR and noted that no indications of significant degradation have been identified in past ASME Code, Section XI, Subsection IWE inspections, and that indications of degradation in inaccessible areas related to the containment moisture barrier and insulated containment liner have been identified and properly addressed. In addition, the licensee has addressed the industry-wide operating experience discussed in IN 2014-07 by revising the IWE program to perform visual inspections of accessible tubing in the PPS. Based on its review, the NRC staff finds that the licensee is properly implementing the ASME Code, Section XI, Subsection IWE program.

In Section 3.2.2 of the LAR, the licensee described the IWL CISI Program and provided an examination schedule from 2000 through 2030, covering the first through seventh 5-year inspection intervals. Two relief requests (RRs), RR-01 (Containment Liner) and RR-02 (Containment Moisture Barrier) (Reference 20), approved by the NRC on April 30, 2009 (Reference 21), were discussed in the LAR. The RRs are related to performance of visual examinations on accessible surface areas of the containment liner and moisture barrier, which have insulation panels attached to the surface. The licensee stated that these reliefs will be implemented during the second 10-year IWE/IWL inspection interval (September 9, 2008 – September 8, 2018) for containment inspections required by ASME Code Section XI. The LAR stated that the results associated with the containment liner first 10-year inspection interval indicated that after removal of 108 insulation panels, degradation of the coating was identified, which required removal and reapplication of the coating. Subsequent ultrasonic and visual examinations did not identify any issues associated with the minimum wall thickness of the liner.

Results for the containment moisture barrier visual examinations performed during the first 10-year inspection interval for 100 percent of the moisture barrier indicated degradation of the barrier in some locations, with no minimum wall thickness violations noted. A recent IWL visual inspection of concrete for the containment building during the fall of 2013 identified several newly reported adverse conditions, which, per ASME Code Section XI, were evaluated and found acceptable by a registered professional engineer. Some of the conditions noted during the inspection appear to be from initial concrete placement, minor rusting, and some pop-outs, which were determined to be cosmetic in nature, with no concrete spalling, exposure, or loss of material associated with the reinforcing steel identified. The licensee noted that none of the conditions identified represented structural concerns and that none impacted the ability of the containment to perform its safety function or the leak-tightness of the containment. The licensee stated in the LAR that the indications found in past inspections were acceptable per the IWL acceptance criteria. Based on its review, the NRC staff finds that the licensee is properly implementing the ASME Code, Section XI, Subsection IWL program.

3.2.5 NRC Conditions In NEI 94-01, Revision 2-A

In the NRC SE dated June 25, 2008 (Reference 10), the NRC staff concluded that the guidance in NEI 94-01, Revision 2 (Reference 8), is acceptable for reference by licensees proposing to amend their TSs with 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 was extended to 15 years. The SE conditions contained in NEI 94-01 Revision 2-A (Reference 5), however, were inadvertently left out of NEI 94-01, Revision 3 (Reference 9), and were not carried forward into the NRC SE for NEI 94-01, Revision 3 (Reference 11). To ensure that licensees acknowledge the limitations and conditions in the NEI 94-01, Revision 2, SE, the NRC staff evaluated whether the licensee adequately addressed these conditions in the LAR.

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. (Reference 10, SE Section 3.1.1.1).

The licensee stated in the LAR that it would be using the definition in NEI 94-01, Revision 2-A, Section 5.0. The requirement is the same in Revision 3-A, Section 5.0. Therefore, the licensee 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. (Reference 10, SE Section 3.1.1.3).

The LAR provided the current schedule of containment inspections in Table 3.2.2-1 for the ASME Code, Section XI, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants," and Table 3.2.2-2 for Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants." The LAR response to Condition 2 also stated that the performance of a pretest visual inspection of accessible interior and exterior surfaces of containment structures and components for evidence of deterioration is completed and documented as a requirement of the HBRSEP2 Containment Integrated Leak Test surveillance procedure. 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. (Reference 10, SE Section 3.1.3).

The licensee described in LAR Section 3.2.2 the inaccessible Class MC areas and the augmented examinations. 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. (Reference 10, SE Section 3.1.4).

The licensee's LAR indicated that HBRSEP2 steam generator replacements were completed in 1984 using the equipment hatch and that there were no planned modifications that would require a Type A test. Therefore, the licensee 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. (Reference 10, SE Section 3.1.1.2).

The licensee's 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 HBRSEP2 as it was not licensed under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

3.2.6 NRC Conditions in NEI 94-01, Revision 3-A

In the NRC SE dated June 8, 2012 (Reference 11), the NRC staff concluded that the guidance in NEI 94-01, Revision 3 (Reference 9), is acceptable for reference by licensees proposing to amend their TSs with 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 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 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 [boiling water reactor] MSIVs [main steam isolation valves]), and those valves with a history of leakage, or 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 routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C combined total, and must be included in a licensee's 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 LAR indicated that the HBRSEP2 post-outage report 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 Type C combined total exceed an administrative limit of 0.5 L_a but be less than the TS acceptance value, then an analysis will be performed, and a corrective action plan prepared, to restore and maintain the leakage summation margin to less than the administrative limit. The LAR also stated that HBRSEP2 will apply the 9-month 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 by administrative limits and understatement contribution adjustments. Therefore, the licensee addressed and satisfied NRC Conditions 1 and 2 of NEI 94-01, Revision 3-A.

3.2.7 Summary of Deterministic Considerations – Structural and Leak-Tight Integrity of the Containment

The NRC staff reviewed the information related to the licensee's proposal to extend 10 CFR Part 50, Appendix J, test intervals, including leakage test results and ASME Code inspection results. The results provided in Section 3.2.4 of the LAR indicated that the previous seven consecutive Type A tests were successful with containment performance leakage rates less than the maximum allowable containment leakage rate of 0.1 percent containment air weight per day (0.75 L_a at P_a). The licensee stated that considerable margin exists between these Type A test results and the TS 5.5.16 limit of 0.75 L_a, which demonstrates a low leakage containment. Therefore, the NRC staff finds that the performance history of Type A tests supports extending the current ILRT interval to 15 years.

Additionally, the NRC staff reviewed the ASME Code, Section XI, Subsection IWE and Subsection IWL inspection information and noted that the examinations have been completed successfully, with no significant indications of degradation. Based on acceptable results of recent ASME Code, Section XI, Subsection IWE and Subsection IWL inspections, the NRC staff finds that there is no evidence to date of significant degradation and that the licensee is adequately implementing its CISI Program to monitor and manage degradation of the containment.

In summary, the NRC staff finds that the licensee adequately implements its containment ILRT program (Type A leakage tests), its CISI Program, and necessary supplementary inspections to periodically examine, monitor, and manage age-related and environmental degradation of the containment. The results of past ILRTs and the containment concrete and liner visual inspections demonstrate acceptable performance and that the structural and leak-tight integrity of the containment is adequately maintained. Thus, the NRC staff concludes that there is reasonable assurance that the structural and leak-tight integrity for the HBRSEP2 containment will continue to be maintained, without undue risk to public health and safety, if the current Type A interval is extended as requested by the licensee in the LAR.

In addition, based on the preceding regulatory and technical evaluations, the NRC staff finds that the licensee has adequately implemented its LLRT program. The results of the recent ILRTs and LLRTs demonstrate acceptable performance and that the structural and leak-tight integrity of the containment structure is adequately managed and will continue to be periodically monitored and managed by the ILRTs and LLRTs. The NRC staff finds that the licensee has addressed the NRC conditions to demonstrate the acceptability of adopting NEI 94-01, Revision 3-A, and the limitations and conditions identified in the NRC staff SE incorporated in NEI 94-01, Revision 2-A, without undue risk to public health and safety. Therefore, the staff concludes that the proposed changes to HBRSEP2 TS 5.5.16, TS SR 3.6.1.1, and TS SR 3.6.2.1 regarding the primary Containment Leakage Rate Testing Program are acceptable from a deterministic perspective. Furthermore, based on the above assessment, the NRC staff concludes that the proposed change meets 10 CFR 50.36 requirements and is, therefore, acceptable.

3.3 Probabilistic Risk Assessment

3.3.1 Background

Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of NEI 94-01, Revision 3-A (Reference 4), states that 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, Revision 3-A, states that the assessment should be performed using the approach and methodology described in Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2-A,[2] "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (Reference 22). 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 SE dated June 25, 2008 (Reference 10), the NRC staff found the methodology in EPRI TR-1009325, Revision 2 (Reference 23), acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied. These conditions, set forth in Section 4.2, "Limitations and Conditions for EPRI Report No. 1009325, Revision 2," stipulate that:

1. The licensee submits documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 (Reference 14), relevant to the ILRT extension application.

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] "Acceptance Guidelines," 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 preexisting containment large leak accident case (i.e., accident case 3b) used by licensees is assigned a value of 100 times the maximum allowable leakage rate (L_a) instead of 35 L_a.

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

[3] The SE 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.

4. A LAR is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance.

3.3.2 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval from 10 years to 15 years. The risk analysis was provided in Attachment 4 to the LAR dated November 19, 2015 (Reference 1).

In Section 1.0, "Purpose," of Attachment 4 to the LAR, the licensee stated that the plant-specific risk assessment follows the guidance in:

- NEI 94-01, Revision 3-A (Reference 4);
- the methodology outlined in EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994 (Reference 24);
- 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" (Reference 17);
- the methodology used for the Calvert Cliffs Nuclear Plant (CCNP) liner corrosion analysis described in a letter to the NRC dated March 27, 2002 (Reference 25); and
- the methodology described in EPRI TR-1009325, Revision 2-A (Reference 22).

In Section 5.2.6, "Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage," of Attachment 4 to the LAR, the licensee stated that the liner corrosion issue was incorporated into the ILRT extension risk evaluation utilizing the CCNP methodology (Reference 25) and considered applicable liner corrosion events through October 2013.

The licensee addressed the four conditions for the use of EPRI TR-1009325, Revision 2 (Reference 23), which are described in Section 4.2 of the associated SE (Reference 10). A summary of how each condition has been met is provided in the sections below.

3.3.3 Technical Adequacy of the Probabilistic Risk Assessment

The first condition stipulates that the licensee submits documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

In RIS 2007-06 (Reference 15), the NRC clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200 (Reference 16) to assess technical adequacy of the PRA used to support risk-informed applications. Revision 2 of RG 1.200 (Reference 14) will be used for all risk-informed applications received after March 2010. In Section 3.2.4.1, "Quality of the PRA," of the SE for EPRI TR-1009325, Revision 2 (Reference 10), the NRC staff stated, in part, that:

> Licensee requests for a permanent extension of the ILRT surveillance interval to 15 years pursuant to NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, will be treated by NRC staff as risk-informed license amendment requests. Consistent with information provided to industry in Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," the NRC staff will expect the licensee's supporting Level 1/LERF PRA to address the technical adequacy requirements of RG 1.200, Revision 1... Any identified deficiencies in addressing this standard shall be assessed further in order to determine any impacts on any proposed decreases to surveillance frequencies. If further revisions to RG 1.200 are issued which endorse additional standards, the NRC staff will evaluate any application referencing NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, to examine if it meets the PRA quality guidance per the RG 1.200 implementation schedule identified by the NRC staff.

In the same section of the SE, the NRC staff stated that Capability Category (CC) I of the ASME and American Nuclear Society (ANS) PRA Standard RA-Sa-2009,[4] "Standard for Level 1/Large Early Release Frequency PRA for NPP [Nuclear Power Plant] Applications," shall be applied as the standard for assessing PRA quality for ILRT extension applications as 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.

<u>Internal Events</u>

In Attachment 1, "Internal Events PRA Quality Statement for Permanent 15-Year ILRT Extension," of Attachment 4 to the LAR, the licensee discussed the technical adequacy of the internal events PRA. In May 2010, a full scope peer review of the HBRSEP2 internal events PRA model (including internal flooding events) was conducted. The peer review was performed using ASME/ANS Standard RA-Sa-2009 and RG 1.200, Revision 2. The peer review identified 25 supporting requirements within the internal events portion of the model that did not meet CC II, as shown in Table 1 of Attachment 1 of Attachment 4 to the LAR. The licensee stated that Table 1 includes discussion of all Facts and Observations (F&Os) within the internal events portion of the model from the peer review along with disposition of those F&Os. The licensee stated in Table 1 that those findings have been dispositioned. The NRC staff reviewed all F&Os associated with the 2010 HBRSEP2, Internal Events PRA model and their disposition and concludes that, based on the disposition provided by the licensee, all supporting requirements associated with those F&Os are adequately dispositioned for the ILRT interval extension request and, therefore, that the internal events model is of sufficient quality to evaluate risk associated with the permanent extension of the Type A containment ILRT frequency for HBRSEP2.

<u>External Events</u>

In Section 3.2.4.2, "Scope of the PRA," of the SE for EPRI TR-1009325, Revision 2 (Reference 10), the NRC staff stated that:

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[4] The SE for EPRI TR-1009325 actually discusses the use of ASME PRA Standard RA-Sa-2003. With the issuance of Revision 2 of RG 1.200, ASME PRA Standard RA-Sa-2009 is now the applicable version.

> 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.

In Section 5.3.1, "Potential Impact from External Events Contribution," of Attachment 4 to the LAR, the licensee performed an analysis of the external events contributions.

The licensee stated in Section 3.7.2, "Summary of Internal Events PRA Quality Statement for Permanent 15-Year ILRT Extension," of the enclosure to the LAR that the HBRSEP2 internal events model was also updated to support its LAR to adopt National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" (2001 Edition), dated September 16, 2013 (Reference 26). The licensee further stated that the HBRSEP2 combined internal events and fire PRA was peer-reviewed by the Westinghouse Owner's Group against the fire PRA standard supporting requirements from Section 4 of the ASME/ANS Standard. Eighteen finding level and nine suggestion level F&Os were identified during the peer review conducted in March 2013. In July 2013, a focused peer review was conducted to evaluate a certain Technical Element based on refinements to approved methodologies as well as updated documentation. During the focused-scope peer review, seven new findings were identified. The licensee stated that Table 2 of Attachment 1 of Attachment 4 to the LAR documents the F&Os associated with both the initial and focused peer reviews. The licensee further stated that all but two findings have been dispositioned, and fire PRA supporting requirements meet CC II or better. While the quality of the licensee's fire PRA model was reviewed in detail by the NRC staff as a part of the HBRSEP2 NFPA-805 application review, the staff reviewed the dispositions provided in Table 2 and the impact assessment for open findings and concludes that given the level of quality needed for this application, the HBRSEP2 fire PRA is of sufficient quality to evaluate the risk associated with the permanent extension of the Type A containment ILRT frequency.

For seismic risk contribution, the licensee stated that the seismic PRA results from the Individual Plant Examination of External Events (IPEEE) Seismic Margins Analysis do not result in an estimate of CDF. Therefore, the licensee used the estimated seismic CDF at plants in the Central and Eastern United States, which included HBRSEP2, documented in 2010 by Generic Issue (GI)-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment" (Reference 27). To assess the high-wind hazard, the licensee used the high-wind PRA results from the IPEEE. The NRC staff agrees that these IPEEE studies, once updated, generally provide an order of magnitude estimate for contribution of the external events as expected for the application. Because the licensee included a quantitative assessment of the contribution of external events by using the fire PRA model, seismic-risk estimates from GI-199, and high-wind risk contribution obtained from IPEEE (Section 5.3.1 of Attachment 4 to the LAR), the staff concludes that the impact of external events on the risk increases associated with extending the Type A containment ILRT frequency is appropriately considered for this application by an order of magnitude estimate.

Based on the above assessments and given the level of PRA quality needed to support a permanent extension of the Type A containment ILRT frequency, the NRC staff concludes that the licensee's internal events PRA model is of sufficient quality for supporting the evaluation of changes to ILRT frequency and that the risk of external events is appropriately considered and, therefore, the NRC staff finds that the first condition of the SE for EPRI TR-1009325, Revision 2 (Reference 10) is met.

3.3.4 Estimated Risk Increase

The second condition of the SE for EPRI TR-1009325, Revision 2 (Reference 10) stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small and consistent with the guidance in RG 1.174 (Reference 17) and the clarification provided in Section 3.2.4.6, "Acceptance Guidelines," of the NRC SE for EPRI TR-1009325, Revision 2 (Reference 10). 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-Roentgen equivalent man (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. Additionally, for plants that rely on containment over-pressure for net positive suction head 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. As discussed further in Section 3.3.6 of this SE, HBRSEP2 does not credit containment over-pressure. Thus, the associated risk metrics include LERF, population dose, and CCFP.

Details of the risk assessment are provided in Attachment 4 to the LAR (Reference 1). The risk impacts are reported for a change in the Type A ILRT test interval from three per-10-years (the test frequency under 10 CFR Part 50, Appendix J, Option A) to once per-15-years (risk impact from baseline), and from once per-10-years to once per-15-years (risk impact from current). The licensee reported the results of the plant-specific risk assessment in Section 5.3.1 of Attachment 4 to the LAR.

The NRC staff draws the following conclusions from the licensee's analysis associated with extending the Type A containment ILRT frequency:

1. The reported increase in LERF for a change in test frequency from three tests in 10 years to one test in 15 years is 3.57E-07 per year. This number includes both internal and external events (i.e., internal fires, seismic events, and high wind) and the impacts from corrosion. This change in internal and external events risk is considered to be "small" (i.e., between 1E-07 per year and 1E-06 per year) per acceptance guidelines in RG 1.174. According to RG 1.174, an assessment of the baseline LERF is required to show that the total LERF is less than 1E-05 per reactor year. The licensee estimated the new total LERF to be 7.33E-06 per year, which is below the total LERF value of 1E-05 per reactor year in RG 1.174.

2. The increase in CCFP due to a change in test frequency from three tests in 10 years to one test in 15 years is 0.829 percent. This value is below the acceptance guideline of 1.5 percentage points for a small increase in CCFP in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2 (Reference 10).

3. Given a change in Type A ILRT frequency from three tests in 10 years to one test in 15 years, the reported increase in the total population dose is 0.20 person-rem per year. The increase in population dose is less than the values associated with a small increase, as provided in EPRI TR-1009325, Revision 2-A (Reference 22), and defined in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2.

Based on the NRC staff evaluation of the licensee's risk assessment results, the NRC staff concludes that the increase in LERF is small and consistent with the acceptance guidelines of RG 1.174. In addition, the increase in the total population dose and the magnitude of the change in the CCFP for the requested change are small and supportive of the proposed change. The defense-in-depth philosophy is maintained, because the independence of barriers will not be degraded as a result 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 NRC staff finds that the second condition of the SE for EPRI TR-1009325, Revision 2 (Reference 10) is met.

3.3.5 Leak Rate for the Large Preexisting Containment Leak Rate Case

The third condition stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2 (Reference 22), acceptable, the average leak rate for the preexisting 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 3.7.1, "Evaluation of Risk Impact," of the LAR (Reference 1), the methodology in EPRI TR-1009325, Revision 2-A, incorporates the use of 100 L_a as the average leak rate for the preexisting containment large leak rate accident case, and this value was used in the plant-specific risk assessment. Accordingly, the NRC staff finds that the third condition of the SE for EPRI TR-1009325, Revision 2 (Reference 10) is met.

3.3.6 Applicability if Containment Over-Pressure is Credited for ECCS Performance

The fourth condition of the SE for EPRI TR-1009325, Revision 2 (Reference 10) stipulates that in instances where containment over-pressure is relied upon for ECCS performance, a LAR is required to be submitted. In Section 3.7.1 of the LAR (Reference 1), the licensee stated that HBRSEP2 does not rely on containment overpressure for ECCS performance. Accordingly, the fourth condition is not applicable.

3.3.7 Probabilistic Risk Assessment Conclusion

Based on the above discussions, the NRC staff concludes that the LAR as supplemented (References 1 and 2) for a permanent extension of the Type A containment ILRT frequency from once in 10 years to once in 15 years for HBRSEP2 is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 changes the SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on March 15, 2016 (81 FR 13841). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Glover, R.M., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission (NRC), "Proposed Amendment to Technical Specification 5.5.16 for the Adoption of Option B of 10 CFR 50, Appendix J for Type B and Type C Testing and the Permanent Change in 10 CFR 50, Appendix J, Integrated Leak Rate Test Interval and Type C Leak Rate Testing Frequency," November 19, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15323A085).

2. Glover, R.M., Duke Energy Progress, Inc., letter to NRC, "Supplement to Proposed Amendment to Technical Specification 5.5.16 for the Adoption of Option B of 10 CFR 50, Appendix J for Type B and Type C Testing and the Permanent Change in 10 CFR 50, Appendix J, Integrated Leak Rate Test Interval and Type C Leak Rate Testing Frequency," August 18, 2016 (ADAMS Accession No. ML16235A187).

3. NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995 (ADAMS Accession No. ML003740058).

4. Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012 (ADAMS Accession No. ML122210254 (package); ML12221A202 (report)).

5. NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," November 19, 2008 (ADAMS Accession No. ML100620847).

6. Mozafari, B.L., NRC, letter to C.S. Hinnant, Carolina Power & Light Company, "Issuance of Amendment No. 169 to Facility Operating License No. DPR-23 Regarding Performance-Bases Containment Integrated Leak Rate Testing - H. B. Robinson Steam Electric Plant, Unit No. 2 (TAC No. M94612), May 28, 1996 (ADAMS Accession No. ML020530639).

7. NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 21, 1995 (ADAMS Accession No. ML11327A025).

8. NEI 94-01, Revision 2, "'Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," August 31, 2007 (ADAMS Accession No. ML072970206).

9. NEI 94-01, Revision 3, "'Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," June 2011 (ADAMS Accession No. ML112920567 (report); ML112920561 (transmittal)).

10. Maxin, M.J., NRC, letter to John C. Butler, NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,' and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, 'Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals' (TAC No. MC9663)," June 25, 2008 (ADAMS Accession No. ML081140105).

11. Bahadur, S., NRC, letter to Biff Bradley, NEI, "Final Safety Evaluation of Nuclear Energy Institute (NEI) Report 94-01, Revision 3, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J' (TAC No. ME2164)," June 8, 2012 (ADAMS Accession No. ML121030286).

12. NRC NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 4.0, April 2012 (ADAMS Accession Nos. ML12100A222 and ML12100A228).

13. Technical Specifications Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-52, Revision 3, "Implement 10 CFR 50, Appendix J, Option B," March 8, 2000 (ADAMS Accession No. ML040400371).

14. NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009 (ADAMS Accession No. ML090410014).

15. NRC Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," March 22, 2007 (ADAMS Accession No. ML070650428).

16. NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, January 2007 (ADAMS Accession No. ML070240001).

17. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011 (ADAMS Accession No. ML100910006).

18. NRC Information Notice 2010-12, "Containment Liner Corrosion," June 18, 2010 (ADAMS Accession No. ML100640449).

19. NRC Information Notice 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," May 5, 2014 (ADAMS Accession No. ML14070A114).

20. Baucom C.T., Progress Energy Carolinas, Inc., letter to NRC, "Response to Request for Additional Information Related to Request for Relief from ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL Requirements for Containment Inspections," January 30, 2009 (ADAMS Accession No. ML090340524).[5]

21. Boyce, T.H., NRC, letter to Eric McCartney, Carolina Power & Light Company, "H. B. Robinson Steam Electric Plant, Unit No. 2 - Request for Relief from ASME Code, Section XI, Subsection IWE Requirements for Containment Inspections (TAC No. MD8509)," April 30, 2009 (ADAMS Accession No. ML091170345).

22. Electric Power Research Institute (EPRI)-1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 2-A, October 2008 (publicly available at www.EPRI.com).

23. EPRI TR-1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 2, Final Report, August 2007 (ADAMS Accession No. ML072970204 (package); ML072970208 (report)).

24. EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," August 1994 (publicly available at www.EPRI.com).

25. Cruse, C.H., Constellation Nuclear, letter to NRC, "Calvert Cliffs Nuclear Plant, Unit No. 1; Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," March 27, 2002 (ADAMS Accession No. ML020920100).

26. Gideon, W.R., Duke Energy Progress, Inc., letter to NRC, "License Amendment Request (LAR) to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)," September 16, 2013 (ADAMS Accession No. ML13267A210 (package)).

[5] As stated in the related NRC safety evaluation (Reference 17), in its request for additional information response submittal dated January 30, 2009, the licensee revised and resubmitted relief requests in their entirety (originally submitted on April 4, 2008). Since the NRC staff's safety evaluation is based on the licensee's January 30, 2009, submittal, this is the reference provided.

27. Hiland, P., NRC, memorandum to Brian W. Sheron, "Safety/Risk Assessment Results for Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants,'" September 2, 2010 (ADAMS Accession No. ML100270582 (package)).

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Date: October 11, 2016

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/

Dennis J. Galvin, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 247 to DPR-23
2. Safety Evaluation

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