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

October 27, 2016

Vice President, Operations
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: EXTENSION OF CONTAINMENT LEAKAGE TESTS FREQUENCY (CAC NO. MF7037)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 191 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1 (RBS). The amendment consists of changes to the technical specifications (TSs) in response to your application dated October 29, 2015, as supplemented by letters dated April 19 and July 27, 2016.

The amendment revises TS 5.5.13, "Primary Containment Leakage Rate Testing Program," by incorporating Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, as the implementation document for the RBS performance-based containment leakage rate testing program. Based on the guidance in NEI 94-01, Revision 3-A, the change allows the RBS Type A test (Integrated Leak Rate Test, or ILRT) frequency to be extended from 120 to 180 months, and the Type C tests (Local Leak Rate Tests, or LLRTs) frequency to be extended from 60 to 75 months. Additionally, the amendment modifies Surveillance Requirement (SR) 3.6.5.1.3 to extend the frequency of the Drywell Bypass Test from 120 to 180 months and revises its allowed extension per SR 3.0.2 from 12 to 9 months.

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

Sincerely,

James Kim

James Kim, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 191 to NPF-47
2. Safety Evaluation

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

ENTERGY GULF STATES LOUISIANA, LLC

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
License No. NPF-47

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee or EOI), dated October 29, 2015, as supplemented by letters dated April 19 and July 27, 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, as amended, 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 license 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, and Paragraph 2.C.(2) of Facility Operating License No. NPF-47 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 191 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen S. Koenick, Acting Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-47 and
Technical Specifications

Date of Issuance: October 27, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of the Facility Operating License No. NPF-47 and 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.

Facility Operating License

Remove	Insert
-3-	-3-

Technical Specifications

Remove	Insert
3.6-61	3.6-61
5.0-16	5.0-16

(3) EOI, pursuant to the Act and 10 CFR Part 70, to receive, possess and to use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(6) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

EOI is authorized to operate the facility at reactor core power levels not in excess of 3091 megawatts thermal (100% rated power) in accordance with the conditions specified herein. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 191 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.3	Verify bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is ≤ 10% of the drywell bypass leakage limit.	--------NOTE------- SR 3.0.2 is not applicable for extensions > 9 months ---------------------- 24 months following 2 consecutive tests with bypass leakage greater than the bypass leakage limit until 2 consecutive tests are less than or equal to the bypass leakage limit <u>AND</u> 48 months following a test with bypass leakage greater than the bypass leakage limit <u>AND</u> 180 months

5.5.11 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

d. Proposed changes that do not meet the criteria of either Specification 5.5.11.b.1 or Specification 5.5.11.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 DELETED

5.5.13 Primary Containment Leakage Rate Testing Program

A program shall be established to implement 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 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, Section 4.1, dated October 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 7.6 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.325% of primary containment air weight per day.

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

The provisions of SR 3.0.2 do not apply to test frequencies specified in the Primary Containment Leakage Rate Testing Program.

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

5.5.14 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

(continued)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191 TO

FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By application dated October 29, 2015 (Reference 1), as supplemented by letters dated April 9, 2016 (Reference 2), and July 27, 2016 (Reference 3), Entergy Operations, Inc. (Entergy, the licensee), requested changes to the technical specifications (TSs) for River Bend Station, Unit 1 (RBS). The supplements dated April 9, 2016, and July 27, 2016, 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) staff's original proposed no significant hazards consideration determination, as published in the *Federal Register* on April 12, 2016 (81 FR 21597).

The proposed changes would revise Technical Specification (TS) 5.5.13, "Primary Containment Leakage Rate Testing Program," by incorporating the Nuclear Energy Institute's (NEI's) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J" (Reference 4), as the implementation document for the RBS performance-based containment leakage rate testing program. Based on the guidance in NEI 94-01, Revision 3-A, the proposed change would allow the RBS Type A test (Integrated Leak Rate Test, or ILRT) frequency to be extended from 120 to 180 months, and the Type C tests (Local Leak Rate Tests, or LLRTs) frequency to be extended from 60 to 75 months. Additionally, the amendment proposes to modify Surveillance Requirement (SR) 3.6.5.1.3 to extend the frequency of the Drywell Bypass Test (DWBT) from 120 to 180 months and to revise its allowed extension per SR 3.0.2 from 12 to 9 months.

In accordance with the guidance in NEI 94-01, Revision 3-A, and the limitations and conditions for NEI's technical report (TR) NEI 94-01, Revision 2 (Reference 5), the proposed changes would permit the performance-based primary containment ILRT maximum interval to be extended from no longer than 120 months to no longer than 180 months provided acceptable performance history and other requirements stated in the report are maintained. In accordance with NEI 94-01, Revision 3-A, the proposed change would also permit the containment isolation

valve LLRT maximum interval to be extended from 60 to 75 months. In addition, TS SR 3.6.5.1.3 would also be revised to extend the frequency for performing the DWBT from 120 to 180 months to remain consistent with the practice of performance during the same refueling outage that the ILRT is performed.

The licensee justified the proposed TS changes by providing historical plant-specific containment leakage testing program results and containment in-service inspection (CISI) program results and supported by a 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 NRC staff reviewed the application and supplemental letters, and provides the following evaluation regarding continued assurance that containment leak-tight integrity and pressure suppression capability will continue to be maintained if the current Type A test and DWBT maximum interval is extended from 120 to 180 months and the Type C maximum test interval is extended from 60 to 75 months.

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, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J of 10 CFR Part 50 includes two options: "Option A - Prescriptive Requirements," and "Option B - Performance-Based Requirements," either of which can be chosen for meeting the requirements of the Appendix. RBS adopted Option B by a license amendment dated December 19, 1995 (Reference 6).

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 the service life of the containment.

Option B of 10 CFR Part 50, Appendix J 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 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 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 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.

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 Commission and endorsed in a regulatory guide.

NEI 94-01, Revisions 2 and 3, have been reviewed by the NRC and approved for use. The final safety evaluation (SE) for Revision 2, issued by letter dated June 25, 2008 (Reference 7), 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 for the Type A test. The final SE for Revision 3, issued by letter dated June 8, 2012 (Reference 8), includes two specific limitations and conditions listed in Section 4.0 of the SE for the Type C test. Revisions 2-A and 3-A of NEI 94-01 include the corresponding SEs.

Section 50.55a, "Codes and standards," of 10 CFR Part 50 contains the CISI requirements that, in conjunction with the requirements of Appendix J, ensure the continued leak-tight and structural integrity of the containment during its service life. Paragraph 10 CFR 50.65(a), "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," states, 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 industrywide operating experience."

Section 10 CFR 50.36, "Technical specifications," states that the TSs include items in five specific categories. These categories include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting condition for operations; (3) SRs; (4) design features; and (5) administrative controls.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Changes

RBS TS 5.5.13 currently states, in part:

> A program shall be established to implement 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 Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, except that the next Type A test performed after the August 15, 1992, Type A test shall be performed no later than April 14, 2008.

The revised RBS TS 5.5.13 would state:

> A program shall be established to implement 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, Section 4.1, dated October 2008.

With this change, RBS will implement NEI 94-01, Revision 3-A, and the limitations and conditions of Section 4.1 of the SE for NEI 94-01, Revision 2-A. NEI 94-01, Revision 3-A, provides that extension of the Type A test interval to 180 months be based on two consecutive successful Type A tests (performance history) and other requirements stated in Section 9.2.3 in 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 where most containment leakage has historically been shown to occur and are expected to continue to be the pathways for a majority of potential primary containment leakage; and 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 were to occur. A license amendment dated March 5, 2003 (Reference 8), approved a one-time 5-year extension to the 10-year ILRT interval for RBS.

NEI 94-01, Revision 3-A, also provides that a Type C test interval up to 75 months (and a limited provision for extension or grace period) could be allowed based in part on two consecutive, successful periodic Type C tests of that valve (performance history).

RBS TS SR 3.5.6.1.3 for verifying acceptable drywell bypass (of the suppression pool) leakage potential currently states:

---NOTE---

SR 3.0.2 is not applicable for extensions > 12 months

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24 months following 2 consecutive tests with bypass leakage greater than the bypass leakage limit until 2 consecutive tests are less than or equal to the bypass leakage limit

<u>AND</u>

48 months following a test with bypass leakage greater than the bypass leakage limit

<u>AND</u>

120 months except that the next drywell leakage rate test performed after the June 24, 1994 test shall be performed no later than June 23, 2009.

The revised RBS TS SR 3.5.6.1.3 would state:

---NOTE---
SR 3.0.2 is not applicable for extensions > 9 months
----

24 months following 2 consecutive tests with bypass leakage greater than the bypass leakage limit until 2 consecutive tests are less than or equal to the bypass leakage limit

<u>AND</u>

48 months following a test with bypass leakage greater than the bypass leakage limit

<u>AND</u>

180 months

By letter dated January 29, 1996 (Reference 10), the NRC issued a license amendment for RBS which allowed an extension of the DWBT frequency from 18 months to 120 months to allow performance during the same refueling outages the ILRT is performed. This was justified, in part, on the basis of acceptable past test results (performance history), and a commitment to perform an assessment of drywell bypass leakage potential each refueling cycle, but with a much less exacting procedure. By performing the DWBT less frequently but in refueling outages that the ILRT is performed, the personnel dose and refueling outage resource impact would be significantly less. That earlier TS change included a performance-based approach such that more frequent testing would be required should a test not meet the acceptance criterion. The licensee submitted a license amendment request (LAR) for a one-time DWBT extension on February 16, 2004 (Reference 11), and a license amendment dated October 15, 2004 (Reference 12), was approved to allow a one-time 5-year extension of the DWBT 10-year interval so that its performance could be maintained paired with the ILRT.

3.2 <u>Deterministic Considerations</u>

RBS is a General Electric boiling-water reactor with a Mark III containment design. The primary containment consists of a reinforced-concrete cylindrical drywell structure, which houses the reactor system, enclosed entirely within a freestanding steel primary containment vessel. The primary containment vessel is itself enclosed by a concrete shield building with an annulus area that is part of the secondary containment. The interior of the drywell is connected to the remaining volume of the primary containment vessel through a suppression pool that is partially inside the drywell between a weir wall and the drywell wall and mostly outside the drywell but still inside the primary containment vessel. This arrangement provides for pressure suppression in the event of a loss-of-coolant accident by channeling steam from the break in the reactor system though vent openings connecting the suppression pool inside and outside the drywell where it is condensed by the cooler pool water. The primary containment vessel provides the "leak tight" barrier against the potential uncontrolled release of fission products during a reactor accident. Technical Specification (TS) 5.5.13 identifies the primary containment allowable leakage rate (L_a) as z 0.325 weight percent of the contained free volume per day at the

calculated maximum primary containment vessel design-basis accident loss-of-coolant accident pressure (P_a) of 7.6 pounds per square inch gauge (psig).

3.2.1 Historical Type A Test ILRT Results

In Reference 1, Section 4.0, Table 4.0-2, the licensee presented the historical results of the Type A ILRT tests as summarized below.

Date	Leakage (Primary Containment Weight % per Day)	Test Pressure (psig)
February 27, 2008	0.1939	8.67
August 15, 1992	0.169	8.7
May 29, 1989	0.090171	8.22

With the maximum allowable or performance ("acceptance" in RBS TS 5.5.13) criterion (L_a) leakage rate being 0.325 weight percent per day at a test pressure of at least 7.6 psig, the testing over the last 27 years showed leakages less than 65 percent of the performance criterion. The allowed value for the first plant startup after testing is 0.75 L_a. The last two consecutive tests showed acceptable leakage and no adverse trend was apparent that would suggest the performance criterion would be exceeded before a next test at the 15-year interval.

3.2.2 Historical Type B and Type C Leak Rate Results

In Reference 1, Section 4.0, Table 4.0-5, the licensee presented the historical results of the combined Type B and Type C (LLRT) test totals as summarized below.

Date	As-Found Minimum Pathway Leakage Rate (sccm)	% of TS 5.5.13 Combined Type B and C Total Acceptance Criterion
March 2015	2,909.5	3.5
March 2013	6,701	8.1
February 2011	4,078	4.9
October 2009	5,034	6.1
February 2008	7,346	8.8
May 2006	7,363	8.9
November 2004	7,574	9.1

These historical results comprise the penetration leakage pathways from primary containment atmosphere into the secondary containment volume where it is collected and filtered before release outside secondary containment. The TS 5.5.13 acceptance criterion for combined Type B and C test total is 0.6 L_a as-found minimum pathway which the application indicated as being 83,061 standard cubic centimeters per minute (sccm). The Type C tests are of the containment isolation valves and are expected to contribute a large majority of the measured combined total. None of these as-found totals exceeded 10 percent of the allowed value and

there was no apparent adverse trend to suggest that extending the maximum Type C testing interval to 75 months for valves not otherwise restricted would result in the acceptance value being exceeded. The allowed value for the first plant startup after testing is 0.6 L_a with the as-left maximum pathway leakages totaled. This is a permissive value to be met before plant restart and is not directly determinative of the acceptability of the containment leakage testing and corrective action programs. The application showed that the as-left totals did not exceed 29 percent of the allowed value.

3.2.3 Historical Secondary Containment Bypass Leak Rate Results

In Reference 1, Section 4.0, Table 4.0-6, the licensee presented the historical results of the combined secondary containment bypass leakage testing, as summarized below. Leakage in these pathways are through those lines that pass from the primary containment to outside secondary containment and would not be collected and filtered before release to the environment. The acceptance criterion for this combined leakage is specified in TS SR 3.6.1.3.9 and converts to a leakage rate of about 14,600 sccm.

Date	As-Found Minimum Pathway Leakage Rate (sccm)	% of TS 3.6.1.3.9 Combined Total Acceptance Criterion
March 2015	974.8	6.7
March 2013	1,113	7.7
February 2011	1,536	10.5
October 2009	1,427	9.8
February 2008	1,803	12.3
May 2006	2,180	14.9
November 2004	1,373	9.4

None of these as-found totals exceeded 15 percent of the allowed value and there was no trend to suggest that extending the maximum Type C testing interval to 75 months for valves not otherwise restricted would result in the acceptance value being exceeded. The allowed value for the first plant startup after testing is evaluated with the as-left maximum pathway leakages totaled similar to that done with the combined Type B and C test results. This would be a permissive value to be met before plant restart and is not directly determinative of the acceptability of the containment leakage testing and corrective action programs. The application showed that the as-left totals did not exceed 68 percent of the allowed value.

3.2.4 Containment Inservice Inspection Program - IWE

The licensee stated that the CISI plan implements the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Subsection IWE, 2001 Edition through the 2003 Addendum. The 10-year CISI interval is divided into three periods. A visual inspection of accessible interior and exterior surfaces of the containment for structural deficiencies that may affect the containment leak tight integrity is performed once each period.

The CISI examinations performed in accordance with Subsection IWE satisfy the general visual examination requirements specified in 10 CFR Part 50 Appendix J, Option B. The identification and evaluation of inaccessible areas are addressed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix). The current CISI interval is summarized below:

Inspection Interval	Inspection Period	Period Start Date	Period End Date	Refuel Outage	Refuel Month/Year
2	3		May 30, 2008	RF-14*1,*2	2-2008*1
3	1	May 31, 2008	June 1, 2011	RF-15*2 RF-16	Fall 2009 Spring 2011
3	2	June 1, 2011	June 1, 2015	RF-17*2 RF-18	Spring 2013 Spring 2015
3	3	June 1, 2015	November 30, 2017	RF-19*2	Spring 2017
4*4	1	December 1, 2017	November 30, 2020	RF-20*2	Spring 2019
4*4	2	December 1, 2020	November 30, 2024	RF-21 RF-22*2,*3	Spring 2021 Spring 2023
4*4	3	December 1, 2024	November 30, 2027	RF-23 RF-24*2	Spring 2025 Spring 2027

*1 Last ILRT performed
*2 CISI examination scheduled/performed
*3 15-year ILRT scheduled
*4 The fourth inspection interval schedule has not been finalized

3.2.5 Historical Drywell Bypass Test Results

In Reference 1, Section 4.0, Table 4.0-3 the licensee presented the historical results of the Drywell Bypass Test, as summarized below. This test verifies that the potential for drywell atmosphere to bypass the suppression pool and over pressurize the primary containment is below the containment response analysis determined value in terms of opening size. This acceptance criterion is invoked in TS SR 3.6.5.1.3 and is identified in RBS Updated Final Safety Analysis Report Section 6.2.1.1.3.4 as currently being 0.81 square feet. TS SR 3.6.5.1.3 also identifies the acceptance criterion for permitting the first startup after performing the test as being less than or equal to 10 percent of the bypass leakage limit value.

Date	Effective Drywell Bypass Flow Area (square feet)
February 2008	0.0208
August 1992	0.0188
June 1989	0.00025

In the LAR for the RBS TS Amendment 87 dated January 29, 1996 (Reference 10), which in part allowed extension of the DWBT from 18 months to 120 months the licensee agreed to perform a qualitative assessment of drywell leak tightness each operating cycle. This assessment included observing the differential pressure decay between the drywell and containment after routine containment venting. This observation, while not nearly as accurate as the actual DWBT, provides data suggesting that the drywell bypass potential is being maintained acceptably low. With the limitations of the method in mind he licensee established an acceptance criterion for this leak tightness assessment pressure drop observation as

0.25 pounds per square inch (psi) or less over a period of 32 minutes. The results since the last DWBT in February 2008 were provided in the Reference 1, Section 4.0, Table 4.0-4 and are summarized below.

Date	Pressure Drop (psi)	Elapsed Time (minutes)
March 2015	0.20	40
February 2013	0.12	45
December 2010	0.08	40
September 2009	0.10	40

The historical DWBT results show drywell bypass potential to have been much less than the current acceptance value and no apparent adverse trend. The assessment observation data provided shows that drywell bypass potential likely continues to be acceptable and that there is no apparent adverse trend to suggest the next DWBT performance at the 15-year interval would not demonstrate acceptably low drywell bypass potential.

3.2.6 Deterministic Considerations Summary

The NRC staff finds that the licensee has adequately implemented its primary containment leakage rate testing program consisting of ILRT and LLRT. Per 50.36(c)(3), surveillance requirements in the TS are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The licensee proposed changes to SR 3.6.5.1.3. As explained above, the revised SR assures that the limiting conditions for operation will be met because 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 LLRT and by the DWBT.

Per 50.36(c)(5), administrative controls in the TS are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. The licensee propose changes to administrative controls TS 5.5.13. As explained in Sections 3.3 and 3.4, the revised TS 5.5.13 assures facility operation in a safe manner because 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 staff SE incorporated in TR NEI 94-01, Revision 2-A (Reference 7), without undue risk to public health and safety.

The licensee has implemented ILRT and LLRT containment leakage rate testing programs and the results of the recent ILRTs and LLRTs demonstrate acceptable performance. The licensee further stated in a formal commitment in Attachment 6 of Reference 1, RBS will continue to perform this qualitative assessment of drywell leak tightness once per operating cycle to support a change to performing the DWBT every 180 months. Therefore, the NRC staff finds that the proposed changes to RBS TS 5.5.13 and TS SR 3.6.5.1.3 regarding the primary containment leakage rate testing program and DWBT are acceptable based on deterministic evaluations.

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

In the NRC's SE dated June 25, 2008 (Reference 7), the staff concluded that the guidance in TR NEI 94-01, Revision 2-A, is acceptable for referencing by licensees proposing to amend their TSs 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 Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," were incorporated and the maximum ILRT interval extended to 15 years. Industry review and familiarization with these changes were extensive during the NEI 94-01 revision process. However, the SE conditions contained in Revision 2-A were inadvertently left out of NEI 94-01, Revision 3, nor carried forward into the NRC staff's SE for Revision 3. To ensure licensees acknowledge the limitations and conditions, in the SE for NEI 94-01, Revision 2, the NRC staff evaluated whether the licensee adequately addressed these conditions in the LAR.

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 that it would be using the definition in NEI 94-01, Revision 3-A, Section 5.0. This approach is acceptable because the definition remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01. Therefore, the licensee addressed and satisfied NRC Condition 1.

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 licensee provided the current schedule of containment inspections in Table 4.0-7 of the LAR. Therefore, the licensee addressed and satisfied NRC Condition 2.

NRC Condition 3

> The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3).

The licensee stated that it conducted general visual examinations of accessible interior and exterior surfaces per its CISI plan, which implements the requirements of ASME Code, Section XI, Subsection IWE and that no areas of potential degradation have been identified in the RBS containment structure. Therefore, the licensee addressed and satisfied NRC Condition 3.

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 licensee indicated that its design change process will address any testing and inspection requirements following future containment modifications to the containment structure as that process evaluates requirements pertaining to the ASME CISI program, 10 CFR Part 50, Appendix J, and ASME Code, Section XI programs. Therefore, the licensee addressed and satisfied NRC Condition 4.

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 stated that RBS acknowledges and accepts this NRC staff position. Therefore, the licensee addressed and satisfied NRC Condition 5.

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 RBS. The licensee was not licensed under 10 CFR Part 52.

3.4 NRC Conditions In NEI 94-01, Revision 3-A

In the NRC's SE dated June 8, 2012 (Reference 8), the staff concluded that the guidance in NEI 94-01, Revision 3-A, is acceptable for referencing by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to two conditions.

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, 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....

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 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 licensee indicated that the RBS post-outage report will include the margin between the Type B and Type C minimum pathway leak rate summation value and secondary containment bypass minimum pathway leakage rate summation value adjusted for understatement and their acceptance criteria. Should the Type B and C combined total exceed an administrative limit of 0.5 L_a or 13,000 sccm for secondary containment bypass leakage but be less than the TS acceptance values, 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 licensee also stated that RBS 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 way the totals are calculated, and will assign some additional margin for monitoring acceptability of results via administrative limits and understatement contribution adjustment. Therefore, the licensee addressed and satisfied NRC Conditions 1 and 2 of NEI 94-01, Revision 3-A.

3.5 Probabilistic Risk Assessment

3.5.1 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) TR-1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," (Reference 13). 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 7), the NRC staff found the methodology in NEI 94-01, Revision 2 (Reference 5), and EPRI TR-1009325, Revision 2 (Reference 13), 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 of the SE for EPRI TR-1009325, Revision 2, stipulate that:

1. The licensee submits documentation indicating that the technical adequacy of its probabilistic risk assessment (PRA) is consistent with the requirements of Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Reference 14), relevant to the ILRT extension application. 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 (Reference 7).

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[1] of the SE for EPRI TR-1009325, Revision 2.

3. The methodology in EPRI TR-1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate for the pre-existing containment large leak rate accident case (accident case 3b) used by the licensee shall be 100 L_a instead of 35 L_a.

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

3.5.2 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A or containment ILRT frequency from 10 to 15 years and extending the frequency of the DWBT from 10 to 15 years. The risk assessment was provided in Attachment 3 of the LAR dated October 29, 2015 (Reference 1). In response to an NRC staff request for additional information (RAI) dated May 20, 2016 (Reference 26), the licensee provided additional information in its letter dated July 27, 2016 (Reference 3).

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[1] Section 4.2 of 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.

In Section 1.1 of Attachment 3 of the Reference 1, the licensee stated that the plant-specific risk assessment follows the guidance in:

1. NEI 94-01, Revision 3-A (Reference 4);[2]

2. EPRI TR-104285 (Reference 15);

3. EPRI Revision 2-A of 1009325;

4. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (Reference 16); and,

5. The methodology for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage (Reference 17).

The licensee requested a one-time ILRT extension on May 14, 2002 (Reference 18), and a one-time DWBT extension on February 16, 2004 (Reference 11), both for performing the test once in 15 years. These one-time extensions were approved in NRC SEs dated March 5, 2003 (Reference 9), and October 15, 2004 (Reference 12), respectively.

In each of the one-time extensions, the licensee used the guidance in EPRI TR-104285, while the updated guidance in EPRI Revision 2-A of 1009325[3] was used in the LAR for the proposed permanent ILRT/DWBT frequency extension. This more recent EPRI document considers the change in population dose, large early release frequency (LERF), and conditional containment failure probability (CCFP), whereas EPRI TR-104285 considered only the change in risk based on the change in population dose.

In Attachment 3 of the application, the risk analysis considered previously approved NRC methods for extending the DWBT frequency for BWR Mark III plants. The methodology outlined for the ILRT risk analysis applies to the containment structure; however, it is generally adopted for the DWBT methodology. The primary difference in the methodology used to evaluate the DWBT (from the ILRT methodology) is in the determination of the conditional probability of an existing drywell leak. The different combinations of drywell and containment leakage sizes were considered in the risk assessment. An additional difference in the methodologies is the assignment of an average drywell leak rate for the pre-existing large leak rate accident case, discussed in Section 3.5.2.3 of this SE.

Attachment 3, Table 4.1-1 of Reference 1 defines accident classes used for the containment Type A ILRT and the DWBT intervals. The containment failure classification of interest for the extended ILRT/DWBT frequency is Class 3, which is described in the application Table 4.1-1, as independent (or random) isolation failures that include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress. Class 3 sequences include core damage sequences in which containment integrity

[2] NEI 94-01, Revision 3-A, added guidance for extending Type C LLRT surveillance intervals beyond 60 months. The guidance for extending Type A ILRT surveillance intervals beyond 10 years is the same as that in Revision 2-A.

[3] EPRI Revision 2-A of 1009325 is also referred to as EPRI 1018243 in the LAR.

is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components as for example, liner breach or bellows leakage, if applicable. Leaks which are classified as a large, early release are designated as "Class 3b" in the application and are evaluated in the risk analysis.

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 has been met is provided in the sections below.

3.5.2.1 Technical Adequacy of the PRA

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.

Internal Events

Consistent with the information provided in Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (Reference 19), the NRC staff uses Revision 2 of RG 1.200 (Reference 14) to assess technical adequacy of the PRA used to support risk-informed applications received after March 2010. In 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 I of the 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 LERF and their distribution among release categories are sufficient for use in the EPRI methodology.

The RBS risk assessment performed for the ILRT/DWBT frequency extension request is based on the current Level 1 and Level 2 PRA models of record, Revision 5. It uses the event tree and fault tree methodology. The LAR Appendix A to Attachment 3, Section A.2.2, describes the RBS PRA maintenance and update process to ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant. According to the LAR, there are no significant plant changes (design or operational practices) that have not yet been incorporated into the PRA models, other than those associated with the diverse and flexible coping strategies (FLEX) initiative (NEI 12-06, Reference 20).

The BWR Owners Group performed in April 2011 a full-scope peer review (as described in the licensee's response (Reference 3) to PRA RAI 6.a of Reference 26) of the RBS internal events PRA model (including internal flooding) using the NEI 05-04 (Reference 21) process. It followed the American Society of Mechanical Engineers/American Nuclear Society ASME/ANS RA-Sa-2009 (Reference 22), as clarified by RG 1.200, Revision 2. Of the applicable supporting requirements, more than 85 percent were satisfied at Capability Category II or greater for RBS with the majority of the not-met supporting requirements related to flooding. The peer review generated 59 findings (i.e., Facts and Observations). The application provided 29 unresolved findings from the peer review including the status of the resolution for each finding and the potential impact of each finding on this application. In response to PRA RAI 6.b, the licensee provided the other 30 findings. The NRC staff reviewed these 59 findings and determined that they have no impact on the ILRT/DWBT application results.

External Events

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, in part, 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.

The licensee performed an analysis of the impact of external events in Section 5.7 of Attachment 3 to the application. The licensee's analysis included contribution for the following in the "other hazards initiator group": seismic, internal fire, internal flood, high winds, external floods and transportation and nearby facility accidents. The licensee's approach was to scale the internal events CDF by a multiplication factor, based on the frequency of the other hazards initiator group, for evaluation of the acceptance criteria discussed in Section 3.5.2.2 of this SE. Although an internal flood event is not an external event, the licensee included it with external events to estimate the multiplication factor. This multiplication factor is applied to the other hazards initiator group in the estimation of the change in LERF.

Scaling of seismic risk does not take into account consideration that some seismic events may increase the size of a pre-existing crack from small to large, thereby adding to the LERF. The licensee addressed this observation in response to PRA RAI 1.a (Reference 3) by performing a sensitivity analysis. The licensee performed a bounding sensitivity analysis that conservatively assumes that the "Class 3a" sequences (i.e., small releases from containment) are binned as a LERF, and determined that the acceptance criteria for this application would still be met.

The seismic CDF reported in Table 5.7-1 of the LAR is one order of magnitude lower than the NRC staff's Generic Issue (GI) 199 study (Reference 23) estimate of 2.5E-5/yr for RBS. In response to PRA RAI 1.b (Reference 3), the licensee explained that the RBS Individual Plant Examination of External Events (IPEEE) seismic study was a reduced-scope seismic margins assessment, and, therefore, the GI 199 result is conservative because it used the safe shutdown earthquake of 0.1 g as the high confidence of low probability of failure (HCLPF) plant-level seismic capacity. The licensee further stated that the licensee's seismic review team reassessed the seismic CDF and concluded that a plant-level HCLPF of 0.3 g was more appropriate for estimating the seismic risk for RBS. This resulted in a revised seismic CDF of 2.5E-6/yr. The licensee's response further noted that using more recent seismic hazard curves for its Fukushima Dai-ichi-related screening evaluation (Reference 24) also resulted in seismic CDFs about an order of magnitude smaller than the GI 199 estimate when HCLPF values were increased up to 0.3 g. While the NRC staff has not reviewed the licensee's seismic CDF

estimates, the staff finds that the licensee's reduced seismic CDF estimates are based on systematic studies which remove conservatism from the GI 199 evaluation and contribute to the margin to the acceptance criteria for this application.

Other external events such as external floods, transportation, and nearby facility accidents had been evaluated in the RBS IPEEE. The licensee concluded that these events are considered negligible in estimation of the external events impact on the ILRT/DWBT extension risk assessment; however, the assessment did not appear to consider the current as-built, as-operated plant. In response to PRA RAI 4 (Reference 3), the licensee assessed these external events for the current plant and concluded that, based on updated information, their contribution to external events risk is still judged to be small and therefore there is no impact on the application. With regard to high winds, the licensee completed an evaluation for RBS in 2004, and, according to the LAR, the high winds CDF estimate is not a significant CDF contributor.

The licensee estimated the internal fires CDF of 2.25E-5/year based on the RBS IPEEE. While the internal fire CDF, as such, represents a source of uncertainty, the application results provided in Section 3.5.2.2 of this SE show that there are margins to the acceptance criteria.

The licensee has estimated the external hazards CDF, which is used in evaluating the acceptance criteria for this application. While there is uncertainty in this estimate, the licensee has performed sensitivity analyses, revisited IPEEE risk insights, and has margin to acceptance criteria. Based on these considerations, the NRC staff finds that the external hazards CDF provides a sufficient estimate to support the risk analysis for this application when taken into account with uncertainties. Furthermore, the licensee has evaluated its internal events PRA against the current ASME PRA standard and Revision 2 of RG 1.200. The NRC staff finds that the licensee has also addressed the findings 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 technical adequacy to support the evaluation of changes to ILRT frequencies. Accordingly, the first condition is met.

3.5.2.2 Estimated Risk Increase

The second condition stipulates that the licensee submits 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 and 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. 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 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 in Section 3.5.2.4 of this SE, RBS does not rely on containment over-pressure for ECCS performance. Thus, the associated risk metrics include LERF, population dose, and CCFP. The results for these risk metrics are reported in the Reference 1, Table 5.7-3.

The reported risk impacts are based on a change in test frequency from three tests in 10 years (the test frequency under 10 CFR 50 Appendix J, Option A) to one test in 15 years. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A ILRT frequency and DWBT frequency:

1. The reported increase in LERF is 3.16E-7/yr in Table 5.7-4 of the LAR (2.51E-8/yr for internal events and 2.9E-7/yr for other hazard groups). This change is considered to be "small" (i.e., between 1E-06/yr and 1E-07/yr) per the acceptance guidelines in RG 1.174. An assessment of total baseline LERF is required to show that the total LERF is less than 1E-05/yr. The LAR estimates the total baseline LERF as 7.04E-7/yr, in Table 5.7-2 of the LAR, and includes the one-in-15 year DWBT/ILRT LERF contribution. Based on this, the total baseline LERF is below the RG 1.174 acceptance guideline of 1E-5/yr for a "small" change.

2. The RBS population dose was derived using the Grand Gulf NUREG/CR-4551 study (Reference 25), following the approach described in EPRI 1009325, Revision 2-A. The reported increase in population dose was 9.09E-02-person-rem/yr, or 0.72 percent of the total population dose for the ILRT/DWBT frequency extension from three-per-10 years to one-per-15 years. The reported increase in population dose is therefore below the values in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2, which are an increase less than the population dose of 1.0-person-rem/yr or 1 percent of the total population dose.

3. The increase in CCFP due to change in test frequency from three in 10 years to one in 15 years was reported to be 1.15 percent, which is below the acceptance guideline of 1.5 percent in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2.

Sensitivity Analyses

Containment Steel Liner Corrosion

EPRI Report No. 1009325, Revision 2-A (Reference 13), requires a sensitivity analysis to assess the impact of assumptions regarding corrosion-induced leakage of steel containments/liners. The methodology calls for a separate, plant-specific assessment of the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended ILRT interval.

The licensee's sensitivity analysis used the Calvert Cliffs methodology (Reference 17) to estimate the risk significance of age-related containment steel liner corrosion large leaks. The method provides an estimate of the likelihood of non-detected containment leakage due to corrosion. This method for assessing ILRT frequency represents a reliability model in which the containment is "as good is as found" and not "as good as new" because there is a likelihood that the ILRT may not detect a containment flaw. Thus, undetected flaws could continue to grow until detected and corrected.

To address flaws, limitation and condition number 3 in the NRC staff's SE for NEI 94-01, Revision 2, Section 4.1, requires the licensee to address the areas of the containment structure potentially subject to degradation, including both accessible and inaccessible areas. This limitation and condition references Section 3.1.3 of that SE which states that plant-specific and generic risk-informed analysis have included specific consideration of degradation in inaccessible areas, and this consideration is based on the availability of that data. In addition, Section 9.2.3.3 of NEI 94-01, Revision 2-A, includes guidance on deficiencies identified during supplemental inspections, and consideration of whether or not the extended ILRT interval should continue to be followed. That is, deficiencies identified during supplemental inspections or at any time between ILRTs should be included in the plant's corrective action program and a determination should be performed to identify the cause of the deficiency and determine appropriate corrective actions. If the containment performance has degraded (considering leak rates), the unit should be removed from an extended ILRT interval, if applicable, and corrective action pursued.

The licensee's upper bound sensitivity analysis is given in the Reference 1, Table 6.1-1, for containment steel liner corrosion. This sensitivity analysis doubled the age-adjusted flaw likelihood every 2 years, increased the likelihood of breach given a flaw in containment (50 percent cylinder-dome and 5 percent basemat), and increased the visual inspection detection failure likelihood (15 percent cylinder-dome and 100 percent basemat). These assumptions were used to estimate the likelihood of non-detected containment leakage. The upper bound increase in LERF was reported in the LAR to be 6.85E-08/yr, which includes the increase due to corrosion, and is small when compared to the RG 1.174 acceptance guidelines. This result is reported as an increase in Class 3b frequency for ILRT/DWBT extension from three-in-10 years to one-in-15 years.

Containment Unit Coolers Not Available

For events in which containment unit coolers operate, drywell leakage was assumed to have no impact on the containment's existing leakage category, since the containment coolers would condense any steam that bypasses the suppression pool. For events in which containment coolers do not operate, the licensee's analysis assumes that any increased DWBT leakage leads to an increased frequency of late containment failure (non-LERF sequences). In response to PRA RAI 2 (Reference 3), which also requested justification for not increasing the frequency of LERF sequences, the licensee performed a sensitivity analysis. The sensitivity analysis removed the non-LERF drywell bypass leakage frequency contribution to the Class 3b estimate. The sensitivity analysis showed that this assumption had a negligible impact on the frequency of Class 3b large leaks.

Based on these 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 and supportive of the application. Accordingly, the second condition is met, conditional on the review of the limitation and condition number 3 from NRC staff's SE for NEI 94-01, Revision 2, Section 4.1, which references Section 3.1.3 of that SE. According to Section 3.1.3 of that SE, risk-informed analysis (both plant-specific and generic) has included specific consideration of degradation in inaccessible areas which is based on the availability of data related to the containment degradation in inaccessible areas.

3.5.2.3 Leak Rate for the Large Pre-Existing Containment Leak Rate Case

The third condition in Section 4.2 of the SE for EPRI TR-1009325, Revision 2, states that the methodology is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order 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 licensee shall be 100 L_a instead of 35 L_a. As noted by the licensee in Attachment 1, Section 5.0, Table 5.0-1, of the application, the RBS analysis used a containment pre-existing containment leak rate of 100 L_a to calculate the increase in population dose for the large leak rate accident case (Class 3b).

Drywell Leakage

In the BWR Mark III containment design the drywell is completely enclosed by the primary containment. As such, drywell leakage does not leak directly to the environment, but is further mitigated by the primary containment. Because of this dual structure, the licensee considered the probability of various drywell and containment leakage combinations. The LAR assumed the drywell leakage had the same categories as the containment (i.e., normal leakage, to have a small pre-existing failure (10 base drywell leakage or DWL_b), or to have a large pre-existing failure (100 DWL_b)). Use of 100 DWL_b for the proposed permanent extension represents an increase from 35 DWL_b used in the RBS one-time DWBT extension. The NRC staff requested the licensee to validate the assumptions that for events in which containment coolers operate, increased drywell leakage from 35 L_a to 100 L_a has no impact on the containment's existing leakage category. In response to PRA RAI 3 (Reference 3), the licensee determined that the DWBT leakage assumption of 100 DWL_b was overly conservative to use. The licensee based its conclusion on a review of the Mark III DWBT data provided in the LAR, and concluded that a drywell leakage rate of 40 DWL_b was justified. The licensee stated that the refinement of drywell leakage values does not change the results of the frequency analysis. This drywell leakage rate is close to 35 DWL_b, which was previously found acceptable by the NRC staff in its SE (Reference 12) for the one-time DWBT extension.

The NRC staff finds that 100 L_a instead of 35 L_a was used for the pre-existing containment large leak rate accident case (i.e., accident Class 3b) in the RBS plant-specific risk assessment for the containment ILRT frequency. Furthermore, the RBS plant-specific risk assessment for the DWBT frequency assumed a drywell leakage rate of 40 DWL_b, which is supported by the licensee's DWBT data analysis and is close to the previously accepted value of 35 DWL_b. Accordingly, the third condition is met.

3.5.2.4 Applicability if Containment Over-pressure is Credited for ECCS Performance

The fourth condition stipulates that a LAR is required in instances where containment over-pressure is relied upon for ECCS performance. As noted by the licensee in Section 5.0, Table 5.0-1, of the application, containment over-pressure is not relied upon for ECCS performance for RBS. Accordingly, the fourth condition does not apply.

3.5.3 PRA Conclusion

Based on the evaluation of the RBS risk assessment above, the NRC staff concludes that the proposed LAR for a permanent extension of the ILRT/DWBT frequency to once in 15 years for

RBS is acceptable since the primary containment leakage rate testing program shall be in accordance with the guidelines contained in the limitation and condition number 3 from NRC staff's SE for NEI 94-01, Revision 2, Section 4.1 in Reference 5, which has been proposed to incorporate into TS 5.5.13 in Entergy's response (Reference 2) to an NRC staff RAI dated March 21, 2016 (Reference 27).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State 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 installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and involve changes to 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 April 12, 2016 (81 FR 21597). 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

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Date: October 27, 2016

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

Sincerely,

/RA/

James Kim, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 191 to NPF-47
2. Safety Evaluation

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