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

August 24, 2015

Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: EXTENSION OF THE INTEGRATED LEAK RATE TESTING TO 15 YEARS (TAC NO. MF4727)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 244 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3 (WF3). This amendment consists of changes to the technical specifications (TSs) in response to your application dated August 28, 2014, as supplemented by letters dated April 15, May 4, and June 18, 2015.

The amendment revises WF3 TS 6.15, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," with a reference to Nuclear Energy Institute (NEI) 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," as the implementation document used by Entergy Operations, Inc., to develop the WF3 performance-based leakage testing program in accordance with Option B, "Performance Based Requirements," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

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

Sincerely,

Michael D. Orenak, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures:
1. Amendment No. 244 to NPF-38
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 244
License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc.(EOI), dated August 28, 2014, as supplemented by letters dated April 15, May 4, and June 18, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.2 of Facility Operating License No. NPF-38 is hereby amended to read as follows:

   2. <u>Technical Specifications and Environmental Protection Plan</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 244, 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. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, 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-38 and
Technical Specifications

Date of Issuance: August 24, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 244

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

Replace the following pages of the Facility Operating License 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
-4-	-4-

Technical Specifications

REMOVE	INSERT
6-24	6-24

or indirectly any control over (i) the facility, (ii) power or energy produced by the facility, or (iii) the licensees of the facility. Further, any rights acquired under this authorization may be exercised only in compliance with and subject to the requirements and restrictions of this operating license, the Atomic Energy Act of 1954, as amended, and the NRC's regulations. For purposes of this condition, the limitations of 10 CFR 50.81, as now in effect and as they may be subsequently amended, are fully applicable to the equity investors and any successors in interest to the equity investors, as long as the license for the facility remains in effect.

(b) Entergy Louisiana, LLC (or its designee) to notify the NRC in writing prior to any change in (i) the terms or conditions of any lease agreements executed as part of the above authorized financial transactions, (ii) any facility operating agreement involving a licensee that is in effect now or will be in effect in the future, or (iii) the existing property insurance coverages for the facility, that would materially alter the representations and conditions, set forth in the staff's Safety Evaluation enclosed to the NRC letter dated September 18, 1989. In addition, Entergy Louisiana, LLC or its designee is required to notify the NRC of any action by equity investors or successors in interest to Entergy Louisiana, LLC that may have an effect on 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. <u>Maximum Power Level</u>

EOI is authorized to operate the facility at reactor core power levels not in excess of 3716 megawatts thermal (100% power) in accordance with the conditions specified herein.

2. <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 244, 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.

PROCESS CONTROL PROGRAM (Continued)

1. Sufficient information to support the change together with the appropriate analyses or evaluation justifying the change(s) and

2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after the approval of the General Manager Plant Operations.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program Manual. This document shall contain:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the level of radioactive effluent control required pursuant to 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose or setpoint calculations.

b. Shall become effective after the approval of the General Manager Plant Operations.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.15 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, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated October, 2008, except that the next Type A test performed after the May 21, 2005 Type A test shall be performed no later than May 20, 2020.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 244 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated August 28, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14241A305), as supplemented by letters dated April 15, May 4, and June 18, 2015 (ADAMS Accession Nos. ML15105A615, ML15124A946, and ML15169A180, respectively), Entergy Operations, Inc. (the licensee), requested changes to the technical specifications (TSs) for Waterford Steam Electric Station, Unit 3 (WF3). The supplements dated April 15, May 4, and June 18, 2015, 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 December 9, 2014 (79 FR 73109).

The proposed change would revise WF3 TS 6.15, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," with a reference to Nuclear Energy Institute (NEI) 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," as the implementation document used by Entergy Operations, Inc., to develop the WF3 performance-based leakage testing program in accordance with Option B, "Performance Based Requirements," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." In accordance with the guidance in NEI 94-01, Revision 2-A, dated October 2008 (ADAMS Accession No. ML100620847), the proposed change would permit the performance-based primary containment Integrated Leak Rate Testing (ILRT), also known as a Type A test, intervals to be extended from no longer than 10 years to no longer than 15 years on a permanent basis, provided acceptable performance history and other requirements discussed in this report are satisfied.

2.0 REGULATORY EVALUATION

2.1 System Description

The containment structure is a concrete-reinforced steel building enclosing the nuclear reactor and associated structures, systems and components, and is designed to contain the escape of radioactive steam or gas under certain conditions, which are spelled out as "Design Basis Accidents" in the Final Safety Analysis Report (FSAR). The WF3 containment structure contains many penetrations, such as the personnel hatch and main steam lines to the reactor turbine that are required to maintain a specified leak rate, as specified in the TSs, to protect the health and safety of the public and environment during a nuclear accident.

2.2 Proposed Changes

In accordance with the guidance in NEI 94-01, Revision 2-A, WF3 proposes to extend the interval for the primary containment ILRT, which is currently required to be performed at 10-year intervals to no longer than 15 years from the last ILRT. The current frequency would require the next ILRT to be performed during the fall 2015 refueling outage (RF). The proposed amendment would allow the next ILRT for WF3 to be performed within 15 years from the last ILRT (i.e., May 21, 2005), as opposed to the current 10-year interval. This would allow successive ILRTs to be performed at 15-year intervals, assuming an acceptable performance history.

WF3 TS 6.15, "Containment Leakage Rate Testing Program," 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 May 12, 1991 Type A test shall be performed no later than May 11, 2006.

The proposed change would revise this portion of TS 6.15 by replacing the reference to Regulatory Guide (RG) 1.163 with a reference to NEI 94-01, Revision 2-A. The date for the next ILRT is also revised. The changes are <u>underlined</u>.

> 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 <u>NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated October, 2008.</u> The next Type A test performed after the <u>May 21, 2005</u> Type A test shall be performed no later than <u>May 20, 2020</u>.

2.3 Regulatory Requirements and Guidance

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

Appendix J to 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.

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the reactor building, including systems and components that penetrate the reactor building, does not exceed the allowable leakage values specified in the TS, and that periodic surveillance of reactor building penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the reactor building and the systems and components penetrating the reactor building. The limitation on reactor building leakage provides assurance that the reactor building would perform its design function following an accident up to, and including, the plant design-basis accident. Appendix J to 10 CFR Part 50 identifies three types of required tests: (1) Type A tests, intended to measure the reactor building overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure containing or leakage limiting boundaries (other than valves) for reactor building penetrations; and (3) Type C tests, intended to measure reactor building isolation valve leakage. Type B and C tests identify the vast majority of potential reactor building leakage paths. Type A tests identify the overall (integrated) reactor building leakage rate and serve to ensure continued leakage integrity of the reactor building structure by evaluating those structural parts of the reactor building not covered by Type B and Type C testing.

The regulations in 10 CFR Part 50, Appendix J, Option B, specify the 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 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 with margin, as specified in the TSs. Option B also requires that a general visual inspection for structural deterioration of the accessible interior and exterior surfaces of the containment, which may affect the containment leak-tight integrity, be conducted prior to each Type A test and at a periodic interval between tests based on the performance of the containment system. The leakage rate test results must not exceed the maximum allowable leakage rate (L_a) with margin, as specified in TS 6.15. 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 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's TS. 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.

The implementation document that is currently referenced in WF3, TS 6.15, is RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (ADAMS Accession No. ML003740058) with one exception. RG 1.163 endorsed NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 21, 1995 (ADAMS Accession No. ML11328A025). 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. NEI 94-01, Revision 0, 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 0, also includes provisions to extend Type C test intervals up to 60 months.

NEI 94-01, Revision 2, has been reviewed by the NRC and approved for use. The final safety evaluation (SE) for NEI 94-01, Revision 2, issued by letter dated June 25, 2008 (ADAMS Accession No. ML081140105), documents the NRC's evaluation and acceptance of Revision 2, subject to six specific limitations and conditions listed in Section 4.1 of the SE, and any licensee proposing to follow Revision 2-A must address the limitations and conditions in the SE. Section 3.1 of the NRC staff's SE provides the staff's position on the adequacy of NEI 94-01 in addressing the performance-based Type A test frequencies. It also addresses the adequacy of pre-test visual inspections, procedures to be used after major modifications to the containment structure, deferral of Type A tests beyond the 15 year interval, and the importance of containment in-service inspection (CISI) mandated by 10 CFR 50.55a, "Codes and standards," in ensuring the leak tightness of containment between the Type A tests.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, dated May 2011 (ADAMS Accession No. ML100910006), describes a risk-informed approach acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. RG 1.174, Revision 2, also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009 (ADAMS Accession No. ML090410014), describes an acceptable approach for determining whether the quality 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. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard, RA-Sa-2009, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," as endorsed by RG 1.200 with clarifications and qualifications, is an acceptable approach for demonstrating the technical adequacy of a PRA used to support licensing actions.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the license amendment request (LAR), to revise the WF3 Containment Leakage Rate Testing Program, against the guidance in Section 4.1, "Limitations and Conditions for NEI TR [Topical Report] 94-01, Revision 2," of the NRC safety evaluation for NEI 94-01, Revision 2, to determine the acceptability of WF3's adoption of NEI 94-01, Revision 2-A, without undue risk to public health and safety.

3.1 Last Two Type A Test Results

To extend the Type A test interval, NEI 94-01, Revision 2-A, provides a guideline that the extension shall be based on two consecutive successful Type A tests (performance history) and other requirements stated in Section 9.2.3 in NEI 94-01, Revision 2-A. The NRC staff reviewed the results of WF3's Type A test performance history to determine if they meet Section 9.2.3 requirements.

In Attachment 1, Section 4.1 of the LAR, the licensee presented the results of the last two Type A tests, summarized in Table 1 below.

TABLE 1: WF3 Last Two Type A Test Results

Test Completion Data	Test Pressure, psig	Performance Leakage Rate (percent of containment air weight per day) As-Left (less than 0.5 percent wt per day for acceptable)
May 12, 1991	44.2	0.0858 (1)
May 21, 2005	45	0.0567 (2)

Note (1): Total Time calculation method
(2): NEI 94-01 defined method

The results in Table 1 indicate that the two most recent consecutive Type A tests at WF3 conducted at test pressure of the calculated peak accident pressure (P_a) were successful with containment performance leakage rate less than the acceptable criterion of 0.5 percent containment air weight per day. The NRC staff finds on the basis that the performance leakage rate for extending Type A test interval is determined by the licensee and is consistent with the definition in Sections 5.0, 9.1.1, and 9.2.3 of NEI 94-01, Revision 2-A (see NRC Condition 1 discussion in Section 3.2.1 of this SE). The performance history of successful completion of two most recent consecutive periodic Type A tests supports extending the current ILRT interval to 15 years.

3.2 NRC Conditions in NEI 94-01, Revision 2-A

In the SE dated June 25, 2008, the NRC staff concluded that the guidance in NEI 94-01, Revision 2, is acceptable for reference by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to the six limitations and conditions noted as in Section 4.1 of the NRC SE for NEI 94-01, Revision 2. The NRC staff evaluated whether the licensee adequately addressed and satisfied these conditions in the LAR, as discussed below.

3.2.1 NRC Condition 1

NRC Condition 1 states:

> 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 ANSIIANS-56.82002. (Refer to SE Section 3.1.1.1).

In a table presented in Attachment 1, Section 4.0 of the LAR, the licensee stated that following NRC approval of this LAR, it will use the definition in Section 5.0 of NEI 94-01, Revision 2-A, for calculating the Type A leakage rate when future WF3 Type A tests are performed. Further, the licensee included this as a formal commitment in Attachment 7 of the LAR, "List of Regulatory Commitments," applicable on a continuing compliance basis following NRC approval of the LAR. On the basis that the licensee has applied the definition in Sections 5.0, 9.1.1, and 9.2.3 of NEI 94-01, Revision 2-A, for calculating the Type A test performance leakage rate to demonstrate leakage integrity and to determine extended Type A test intervals, the NRC staff finds that the licensee has adequately addressed Condition 1.

3.2.2 NRC Condition 2

NRC Condition 2 states:

> 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).

NEI 94-01, Section 9.2.3.2, "Supplemental Inspection Requirements," states, in part, that in order "to provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval of the Type A test is extended to 15 years."

In the supplement dated June 18, 2015, the licensee stated, in part, that "preventative maintenance tasks exist to perform periodic general inspections of the accessible interior and exterior surfaces of the containment vessel." The licensee provided a schedule for a typical 15-year interval between the last Type A test in 2005 (RF13) and the proposed next Type A test in 2020 (RF23), including the inservice inspection intervals and inspection periods, and the corresponding refueling outages.

On the basis that the licensee's schedule of general visual examinations results in three examinations of containment vessel surfaces between Type A tests and one examination immediately prior to the Type A test, the NRC staff concludes that the licensee's inspection schedule plan meets the general visual examination requirements in Section 9.2.3.2 of NEI 94-01, Revision 2-A, and Condition 2 in the NRC staff SE for NEI 94 01, Revision 2.

3.2.3 NRC Condition 3

NRC Condition 3 states:

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

In Attachment 1, Section 4.0 of the LAR, the licensee stated that general visual examinations of accessible interior and exterior surfaces of the containment vessel are conducted in accordance with the WF3 CISI program, which implements the requirements of the ASME Boiler and Pressure Vessel Code (Code), Section XI, Subsection IWE, as required by 10 CFR 50.55a(g).

In Attachment 1, Section 4.4 of the LAR, the licensee stated, in part, that "abnormal degradation of the primary containment structure identified during the conduct of IWE program examinations or at other times is entered into the corrective action program for evaluation to determine the cause of the degradation and to initiate appropriate corrective actions; and degradation of the Moisture Barrier was previously identified during the first period of the second interval (RF10) and entered into the corrective action program. Inspection of the containment [steel surface] below this area of the moisture barrier identified an area of degradation that was added into the IWE program as an augmented examination. In addition to the requirements of Table IWE-2500-1, Owner Elected Examinations were performed every outage from RF10 through RF16. Due to the degradation remaining essentially unchanged over this time period, these areas were evaluated to no longer require augmented examinations...."

The licensee indicated in Section 4.0 of the LAR that there are currently no primary containment surface areas that require augmented examinations in accordance with ASME Section XI, IWE-1240.

In the supplement dated June 18, 2015, the licensee revised Table 4-2, "IWE Inspections," of the LAR. The following information relative to the examinations of inner and outer moisture barriers located at the interface of the containment vessel wall and the concrete floor was included in Table 4-2:

a) In 2003, VT-3 examinations of the inner moisture barrier revealed 13 locations where the moisture barrier has failed by a combination of tearing and cracking. These damaged sections were removed and replaced with new sealant.

b) In 2008, the inner and outer moisture barrier sections MB-01 through MB-15 were inspected in RF15. All sections were satisfactory with the exception of Sections MB-02, -03, -05, and -06 which revealed signs of age related degradation and mechanical damage which required repair.

c) In 2009, the inspection of the inner and outer moisture barrier Sections MB-01 through MB-15 were performed in RF16 with satisfactory results.

d) In 2012, during RF18, the inspection of the inner moisture barrier from 0 degrees and 138 degrees azimuth location was performed and water was found standing on the moisture barrier between the 30 degrees and 70 degrees azimuth location. This finding was attributed to the steam generator replacement activities. Three 18 inch x 18 inch moisture barrier sections were removed and the steel vessel surface area examined at the 30 degrees, 42 degrees, and 70 degrees azimuth locations to assure no active degradation was present. After replacement of these sections of the moisture barrier, an examination of the repaired moisture barrier areas were performed, and the examination results were satisfactory.

e) In 2014, the inner moisture barrier Sections MB-02 through MB-11 were inspected in RF19 with satisfactory results.

In the supplement dated June 18, 2015, the licensee stated that a review of the containment basemat monitoring program history since 1997, did not reveal any instances where conditions

existed in accessible areas that could indicate the presence of or result in degraded conditions in inaccessible areas.

Based on the above information regarding the IWE examinations of the WF3 containment structure and the WF3 operating experience to date, no conditions have been identified that would indicate the presence of any significant degraded condition in the accessible or inaccessible areas affecting the leak tightness or structural integrity of the WF3 containment structure. In addition, the WF3 CISI program contains requirements to evaluate the acceptability of the inaccessible areas if such conditions were identified, in accordance with 10 CFR 50.55a(b)(2)(ix)(A). As such, the NRC staff concludes that the licensee has adequately addressed the intent of Condition 3 of the NRC staff SE for NEI 94-01, Revision 2.

3.2.4 NRC Condition 4

NRC Condition 4 states:

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

In a table presented in Attachment 1, Section 4.0 of the LAR, the licensee stated that it had replaced the WF3 steam generators during December 2012 for which the containment structure required modifications. A snoop (bubble) test of the affected areas of the containment liner was performed with satisfactory results in RF18 in lieu of a Type A test post modification. By letters dated January 4, 2012 (ADAMS Accession No. ML113330137) and April 3, 2012 (ADAMS Accession No. ML120580756), the NRC staff authorized this one-time alternative for the WF3 third 10-year inservice inspection interval.

Furthermore, the licensee stated in the LAR that (1) no modifications that require a Type A test are planned prior to RF23, when the next Type A tests will be performed in accordance with this proposed change; (2) any unplanned modifications to the containment vessel prior to the next scheduled Type A test would be subject to the special testing requirements of Section IV.A of 10 CFR 50, Appendix J; (3) there have been no pressure or temperature excursions in the containment vessel which could have adversely affected its integrity; and (4) there is no anticipated addition or removal of plant hardware within the containment vessel which could affect its leak-tightness.

Based on the above, the NRC staff concludes that the licensee's program will implement the staff's position with regard to post-repair pressure testing following major and minor containment repairs and modifications, as explained in Section 3.1.4 of the NRC staff SE for NEI 94-01, Revision 2. Therefore, the staff concludes that the licensee has adequately addressed Condition 4 of the NRC staff SE for NEI 94-01, Revision 2.

3.2.5 NRC Condition 5

NRC Condition 5 states:

> 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, in part, in the table presented in Attachment 1, Section 4.0 of the LAR that it "acknowledges and accepts the NRC staff position [in Condition 5], as communicated to the nuclear industry in Regulatory Issue Summary (RIS) 2008-27, dated December 8, 2008."

The licensee has, thus, acknowledged and accepted the NRC staff position, with regard to extending the Type A test intervals beyond the approved upper bound limit of 15 years, in Condition 5 and clarified in RIS 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50" (ADAMS Accession No. ML080020394). Accordingly, the NRC staff finds that the licensee has confirmed its understanding that any extension of the Type A test interval beyond the upper-bound performance-based limit of 15 years should be infrequent and should be requested only for compelling reasons, and that the staff will implement the position in RIS 2008-27 in reviewing such license amendment requests. Therefore, the staff finds that the licensee has adequately addressed Condition 5 in its LAR.

3.2.6 NRC Condition 6

NRC Condition 6 states:

> 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 [Electric Power Research Institute] Report No. 1009325, Revision 2, including the use of past ILRT data.

The licensee stated in the table presented in Attachment 1, Section 4.0 of the LAR that this condition is not applicable since WF3 is not licensed to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The NRC staff finds that WF3 is currently an operating reactor licensed under 10 CFR Part 50, and therefore, Condition 6 does not apply.

3.2.7 Conclusion for the NEI 94-01, Revision 2, Conditions

Based on the above evaluation of each condition, the NRC staff determined that the licensee has adequately addressed and satisfied the six conditions in Section 4.1 of the NRC SE for NEI 94-01, Revision 2. Therefore, the staff finds it acceptable for WF3 to adopt NEI 94-01, Revision 2-A, as the implementation document in its TS 6.15, "Containment Leakage Rate Testing Program."

3.3 <u>Type B and Type C Testing Program</u>

The licensee described its Type B and Type C Testing Program in Attachment 1, Section 4.2 of the LAR. The licensee stated that its Appendix J, Type B and Type C testing program, consists of local leak rate testing of penetrations with a resilient seal, expansion bellows double gasketed

manways, hatches and flanges, and containment isolation valves that serve as a barrier to the release of the post-accident containment atmosphere.

In Attachment 1, Section 4.2 of the LAR, the licensee provided the as-left maximum and the as-found minimum pathway leakage rates for the last two refueling outages as shown in the table below. The licensee indicated that the leakage rate values were well below the acceptance criterion of combined Type B and Type C leakage rate of 630,000 standard cubic centimeters per minute (sccm).

Refueling Outage (Date)	Leakage	Value, in sccm
RF18 (2012-2013)	As-Found Minimum Pathway Leakage	52,520
	As-Left Maximum Pathway Leakage	48,205
RF19 (2014)	As-Found Minimum Pathway Leakage	56,849
	As-Left Maximum Pathway Leakage	69,947

The licensee stated that industry experience has shown that Type B and Type C tests can identify the vast majority (over 95 percent) of all potential primary containment leakage paths. The licensee stated, in part, that this LAR "adopts the guidance in NEI 94-01, Revision 2-A, in place of NEI 94-01, Revision 0, but otherwise does not affect the scope, performance or scheduling of Type B or Type C tests. Type B and Type C testing will continue to provide a high degree of assurance that primary containment integrity is maintained."

In the supplement dated May 4, 2015, the licensee stated that 85 percent of the total number of Type B tested components are on a 120-month extended performance base test interval. In addition, in the letter dated June 18, 2015, the licensee stated that WF3 has 95 Type C tested components; which 70 components are eligible to be placed on a 60-month extended interval based on component performance history. Currently, 46 of the eligible 70 of the Type C tested components are on a 60-month extended performance-based test interval.

Based on the information discussed above, the NRC staff finds that (1) the combined leakage of Type B and Type C tests for the last two refueling outages is well below the acceptance limit; (2) a majority of mechanical and electrical penetrations that are subject to performance based Type B or Type C tests are on or could be on the maximum allowed performance based interval of 120 months or 60 months, as applicable, which demonstrates good performance of Type B and Type C penetrations at WF3; (3) there is reasonable assurance that the licensee is effectively implementing its Type B and Type C testing program under Option B of Appendix J to 10 CFR Part 50; and (4) if the current ILRT interval is extended to 15 years, the licensee will continue to appropriately implement Type B and Type C testing in accordance with NRC regulations and the limitations and conditions identified in the staff's SE incorporated into NEI 94-01, Revision 2-A. The NRC staff concludes that the integrity of the containment pressure boundary penetrations (including access hatches and airlocks) and isolation valves are effectively monitored through Type B and Type C testing, as required by 10 CFR Part 50, Appendix J and the implementation document referenced in the WF3 TS.

3.4 Containment In-Service Inspection Program

In Attachment 1, Section 4.0 of the LAR, the licensee stated that general visual examinations of accessible interior and exterior surfaces of the containment vessel for structural problems are conducted in accordance with the WF3 CISI program and schedule, which implement the requirements of the ASME Code, Section XI, Subsection IWE, as required by 10 CFR 50.55a(g).

The operating experience relative to the moisture barrier at the interface of the concrete floor and the containment vessel wall is discussed in Section 3.3.3 of this SE and will not be repeated here.

In the supplement dated May 4, 2015, the licensee stated that in May 2005 (RF13) and in November 2009 (RF16), a general visual inspection of the accessible areas of the inside surface of the containment vessel was performed in accordance with ASME Code, Section XI, Subsection IWE and the inspection results reflect compliance with the containment vessel structural integrity requirements. The licensee also stated that all accessible areas of the outer surface of the containment vessel were examined from the annulus area and no discrepancies were found. Furthermore, in the supplement dated May 4, 2015, the licensee provided information regarding the recent visual inspection of the containment vessel inner and outer surfaces performed in RF18. The licensee stated that the inspection results were satisfactory except for (1) minor areas of rust, with no pitting or loss of thickness, at the containment vessel dome outer surface; and (2) active corrosion at the lug weld to the containment vessel wall, adjacent to penetration number 36, with no pitting or loss of material in the containment vessel wall area.

In the supplement dated June 18, 2015, the licensee stated the following regarding the WF3 monitoring of the Nuclear Plant Island Structure (NPIS) reinforced concrete foundation basemat:

a) The Steel Containment Vessel (SCV) is a low leakage rate free standing steel pressure shell, completely enclosed by the concrete shield structure, with an annular space provided between the walls and domes of each structure to permit construction, operations, and inservice inspection. The SCV is rigidly supported on a concrete base that was placed after the cylindrical shell and the ellipsoidal bottom had been constructed and post weld heat treated. The containment vessel, shield building, reactor auxiliary building, and fuel handling building are supported on a common foundation mat. Concrete floor fill was placed above the ellipsoidal shell bottom of the SCV after the vessel had been post weld heat treated, to anchor the vessel. All components and framing inside the SCV are supported on the concrete floor fill.

b) Although portions of the NPIS basemat are inaccessible for visual inspection, Section 2.5.4.13.4 of the WF3 FSAR, and WF3 TS 6.8.4(e) require monitoring of the NPIS on a continuous basis. The NPIS monitoring program provides for data collection and trending, and the methods for an assessment of the NPIS common foundation basemat structural integrity through measurement of the settlement, and measurement of changes in width of 15 instrumented basemat cracks, on an 18-month frequency.

c) A review of the base mat inspections dating back to 1997 revealed that all of the acceptance criteria were met with satisfactory results.

In the supplement dated June 18, 2015, the licensee revised Table 4-1 of the LAR and provided the summary of the results of structural inspections of the WF3 shield building performed in 1988, 1991, 1995, and 2005. The licensee indicated that the structural integrity inspection of the inside and outside of the shield building is performed prior to any integrated leak rate test. As it is indicated in Table 4-1, the inspection of interior and exterior concrete surfaces of WF3 shield building did not reveal any significant degradation.

In the supplement dated May 4, 2015, the licensee stated, that based on a review of design-basis documents and associated controlled drawings, (1) WF3 does not have any components that should be added to the CISI program equivalent to the items discussed in NRC Information Notice (IN) 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," dated May 5, 2014 (ADAMS Accession No. ML14070A114) and (2) there are no channels installed to encompass the welds in the ellipsoidal bottom head of the steel containment vessel with associated pressurization lines/tubing/valves.

As indicated in Table 4-2, provided in the supplement dated June 18, 2015, VT-3 examinations of the coatings on the interior of the containment vessel, performed in 2003, found areas with flaking, peeling, blistering, and discoloration of the painting. These locations were repaired and reinspected in RF12 with satisfactory results. Also, in the supplement dated May 4, 2015, the licensee provided information regarding the recent inspection of the containment vessel coating performed in May 2014 (RF19). The licensee stated that coating damage in some areas of the containment vessel wall, dome, and polar crane ring girder were identified; but rusting of the substrate was not observed. The licensee initiated a condition report to document the identified coating damaged areas.

Based on the results of the containment vessel inspections, including the inspection of the moisture barrier discussed in Section 3.3.3 of this SE, and the results of the shield building inspections and the containment vessel coating inspections discussed above, the NRC staff finds that there has not been evidence, to date, of significant degradation of the WF3 containment vessel, and the degradations noted have been entered into the WF3 corrective action program, and appropriately managed and/or corrected. Based on the above evaluation, the NRC staff finds that there is reasonable assurance that the licensee is adequately implementing the WF3 CISI program to monitor and manage age-related degradation of the WF3 containment vessel.

3.5 Probabilistic Risk Assessment

3.5.1 Background

NEI 94-01, Revision 2, Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. NEI 94-01, Revision 2, Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," states, in part, that "the assessment should be

performed using the approach and methodology described in EPRI Report 1009325, Rev 2-A[1], "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval."

In the SE of NEI 94-01, Revision 2, dated June 25, 2008, the NRC staff found the methodology in EPRI Report No. 1009325, Revision 2, 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 of NEI 94-01, Revision 2 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 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[2] of the SE for EPRI Report No. 1009325, Revision 2.

3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable provided the average leak rate for the pre-existing containment large leak accident case (i.e., accident case 3b) used by licensees is assigned a value of 100 L_a instead of 35 L_a.

4. 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 plant-specific risk analysis for extending the Type A containment ILRT interval from 10 years to 15 years. The risk analysis is discussed in Attachment 1, Section 4.5, "Plant-Specific Confirmatory Analysis," of the LAR and the detailed analysis is included as Attachment 6, "Risk Analysis." A revised detailed risk analysis was provided by the licensee in Attachment 3, "Calculation, Waterford 3 Evaluation of Risk Significance of an ILRT Extension," to the supplement dated May 4, 2015.

The licensee stated in the LAR that the plant-specific risk analysis followed the guidance in NEI 94-01, Revision 2-A, the methodology in EPRI Report No. 1009325, Revision 2-A, and the guidance in RG 1.174 on the use of PRA. The licensee also stated that it used the methodology used for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval. The licensee stated that its Level 2 PRA model includes internal events and that the dominant external events were evaluated using the information from its Individual Plant Examination of External Events (IPEEE) and its fire PRA. The original LAR did not include an

[1] It should be noted that EPRI Report No. 1009325, Revision 2-A, is also identified as EPRI Report No. 1018243.

[2] The safety evaluation 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.

evaluation of internal flooding of the plant. In the supplement dated May 4, 2015, the licensee revised its baseline PRA results to include evaluation of internal flooding.

The licensee addressed each of the four conditions for the use of EPRI Report No. 1009325, Revision 2, which are listed in Section 4.2 of the NRC SE for EPRI Report No. 1009325, Revision 2. Summaries of how each condition has been met are provided in the following sections.

3.5.2.1 Technical Adequacy of the PRA

The first condition in Section 4.2 of the SE for EPRI Report No. 1009325, Revision 2, stipulates that the licensee submit documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (ADAMS Accession No. ML070650428), clarified that the NRC staff will use Revision 1 of RG 1.200 (ADAMS Accession No. ML070240001) to assess the technical adequacy of PRAs used to support risk-informed applications received after December 2007. Consistent with the wording in RIS 2007-06, all risk-informed applications received after March 2010 will be assessed using Revision 2 of RG 1.200. In Section 3.2.4.1 of the SE for EPRI Report No. 1009325, Revision 2, 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 I of the ASME/ANS PRA standard 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.

Attachment 1, Section 4.5.2 of the LAR discusses the technical adequacy of the WF3 PRA. The LAR stated that the WF3 internal events PRA model was peer reviewed using the guidance in Revision 1 of RG 1.200. In the request for additional information (RAI) by letter dated February 18, 2015 (ADAMS Accession No. ML15033A422), the NRC staff requested that the licensee identify any gaps between the WF3 internal events PRA model used to develop the LAR and the guidance in RG 1.200, Revision 2. The NRC staff also requested that the licensee identify any supporting requirements for which Capability Category I of the 2009 ASME/ANS PRA

Standard was not met. In the supplement dated May 4, 2015, the licensee stated that Revision 4 of the WF3 internal events PRA model was used in preparation of the LAR and that Revision 4 was peer reviewed using the guidance in RG 1.200, Revision 1. The licensee also stated that the peer review found that the PRA met 90 percent of the supporting requirements for Capability Category I or better. The licensee performed a gap assessment to determine if the results of the peer review would have changed if the peer review had been performed using Revision 2 of RG 1.200. The gap assessment determined that no additional findings would have been issued, but two suggestion level facts and observations (F&Os) would have been considered findings.

Following completion of the peer review and submittal of the LAR, the WF3 internal events PRA model was updated to Revision 5. The licensee stated that no changes in methods were associated with the model update, but that most of the findings from the peer review of Revision 4 were addressed in Revision 5 of the WF3 internal events PRA model. The licensee documented the risk analysis results using Revision 5 of the WF3 PRA model as a sensitivity analysis in its response to RAI 1 in the supplement dated May 4, 2015. The sensitivity analysis showed that the risk results increased slightly, but that the risk acceptance guidelines applicable to this LAR were not exceeded.

The licensee stated that any supporting requirements for which Capability Category I of the 2009 ASME/ANS PRA Standard was not met were documented as findings. The licensee assessed the impact of each of the finding level F&Os on this LAR and documented the results in Attachment 2, "Internal Event PRA Peer Review – Facts and Observations (Findings Only)," to the supplement dated May 4, 2015. The licensee also stated that while most of the open F&Os were addressed in Revision 5 of the WF3 PRA model update, nine F&Os related to the internal events model and eight F&Os related to the internal flooding model were not fully addressed by Revision 5 of the WF3 PRA model. The licensee identified two F&Os related to the internal events model and two F&Os related to the internal flooding model that could impact this LAR. The licensee performed an additional sensitivity analysis to double the internal flooding contribution to CDF in order to bound the impact of the four F&Os that had the potential to impact this LAR. The sensitivity analysis showed that the risk acceptance guidelines applicable to this LAR, as discussed in Section 3.5.2.2 of this SE, were not exceeded. The NRC staff reviewed the licensee's assessment of the impact of the findings that were not addressed by Revision 5 of the WF3 PRA. Based on a review of the open F&Os; the sensitivity analysis performed by the licensee; and the margin between the reported risk values and the acceptance guidelines, the NRC staff finds that resolution of the open F&Os would be unlikely to impact the risk results such that the risk acceptance guidelines would be exceeded.

In Section 3.2.4.2, "Scope of the PRA," of the SE for NEI 94-01, Revision 2, and EPRI Report No. 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."

In Attachment 1, Section 4.5.2 of the LAR, the licensee stated that the WF3 PRA model includes Level 2 and LERF models for internally initiated events. The licensee also stated that the severe accident sequences have been mapped to their radiological release end states. The original risk assessment of internal and external events used to support this licensing action is presented in Attachment 6 of the original LAR dated August 28, 2014, but was revised in Attachment 3 to the supplement dated May 4, 2015. The original risk analysis included evaluation of internal events, fire, and seismic events, but did not include evaluation of internal flooding. In the supplement dated May 4, 2015, the licensee revised its risk analysis to include evaluation of internal flooding. The licensee noted that its internal flooding model was peer reviewed along with its internal events model, but that the internal flooding model did not include all Level 2 end states. The licensee also noted that the internal flooding cut sets were similar to those for the transient initiating event (i.e., involved a similar plant response). Therefore, the licensee assumed that half of the CDF contribution would contribute to the INTACT end state and the other half would contribute to the LATE end state, consistent with the end state binning for the transient initiating event. Based on the licensee's reported similarity between internal flooding and transient cut sets, the NRC staff finds that this is a reasonable approach to assigning end states to flooding sequences and therefore provides an appropriate estimate of the risk from internal flooding.

In Section 5.3, "Potential Impacts from External Events," of the risk assessment calculation in Attachment 3 to the supplement dated May 4, 2015, the licensee stated that the impact of external events was evaluated using the information from the IPEEE. Seismic and fire events were considered to be the most limiting for WF3 due to frequency of occurrence and potential impact on the plant and therefore, they are assumed to bound the external events risk contribution for the plant. The licensee also stated that seismic events were addressed using a seismic margins analysis from the IPEEE in conjunction with the updated internal events model. The licensee further stated that the contribution from fire events was calculated using the WF3 fire PRA model.

In Section 4.5.2 of the LAR, the licensee stated that the WF3 fire PRA model was peer reviewed against Sections 2 and 3 of the ASME/ANS PRA Standard. By letter dated February 18, 2015, the NRC staff requested that the licensee clarify whether the WF3 fire PRA was peer reviewed in accordance with Sections 2 and 3 of Part 4, "Requirements for Fire At-Power PRA," of the 2009 ASME/ANS PRA Standard. In the supplement dated May 4, 2015, the licensee stated that its fire PRA was peer reviewed against Part 4 of the 2009 ASME/ANS PRA Standard and the internal events PRA model, used as an input to the fire PRA model, was peer reviewed against Sections 2 and 3 of the ASME/ANS PRA Standard. The NRC staffs finds the licensee's clarification acceptable because the WF3 internal events and fire PRA models were peer reviewed in accordance with the appropriate sections of the ASME/ANS PRA Standard.

The NRC staff reviewed the LAR and the supplement dated May 4, 2015, to determine the technical adequacy of the WF3 PRA model used for this application. Given that the licensee evaluated its PRA against Revision 1 of RG 1.200 and the ASME/ANS PRA standard;

performed a gap assessment of its internal events PRA against Revision 2 of RG 1.200; evaluated the findings developed during the reviews of its PRA for applicability to the ILRT extension; and evaluated the risk contribution from external hazards, the NRC staff concludes that the PRA model used by the licensee is of sufficient technical adequacy to support the evaluation of changes to ILRT frequencies. Accordingly, the first condition in Section 4.2 of the SE for EPRI Report - 1009325 is met.

3.5.2.2 Estimated Risk Increase

The second condition in Section 4.2 of the SE for EPRI Report – 1009325, Revision 2, 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 and the clarification provided in Section 3.2.4.6 of the NRC SE for NEI 94-01, 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 one 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 percent. Furthermore, since WF3 does not rely on containment overpressure for ECCS net positive suction head, the relevant risk metric for the impact of the ILRT extension is LERF, since the Type A test does not generally impact CDF. LERF shall be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. Thus, the associated risk metrics include: LERF, population dose, and CCFP.

The licensee reported the results of the plant-specific risk assessment in its response to RAI 1 in the supplement dated May 4, 2015, and in the LAR, Attachment 1, Section 4.5.3, as revised by the supplement dated May 4, 2015. Details of the risk assessment are provided in Attachment 3 to the supplement dated May 4, 2015. The reported risk impacts are risk impact from baseline, which estimates the impact of 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 and risk impact from current, which estimates the impact of a change from one test in 10 years to one test in 15 years. The following conclusions can be drawn based on the licensee's analysis associated with extending the Type A ILRT frequency:

1. The reported increase in LERF for a change in test frequency from one test in 10 years to one test in 15 years is 2.35×10^{-8} per year. The reported increase in LERF for a change in test frequency from three tests in 10 years to one test in 15 years is 5.64×10^{-8} per year. These numbers include the evaluation of risk from both internal and external events. These changes in internal and external events risk are considered to be "very small" (i.e., less than 1×10^{-7} per year) per the acceptance guidelines in RG 1.174. According to RG 1.174, an assessment of baseline LERF is required to show that the total LERF is less than 1×10^{-5} per reactor year. In Attachment 3 to the supplement dated May 4, 2015, the licensee estimated the total baseline LERF to be 5.31×10^{-7} per year. Thus, the new total LERF, given the increase in ILRT interval, would be approximately 5.5×10^{-7} per year, which is below the total LERF value of 1.0×10^{-5} per reactor year in RG 1.174.

In Section 5.0 of Attachment 3 to the supplement dated May 4, 2015, the licensee presented sensitivity analyses to gain an understanding of the sensitivity of the results to key parameters in the corrosion analysis. The reported increase in total LERF for a change in test frequency from three tests in 10 years to one test in 15 years is 5.69×10^{-8} per year according to the sensitivity analysis performed.

2. The reported increase in total population dose from one in 10 years to one in 15 years is 2.01×10^{-2} person-rem per year, or 0.006 percent of the total population dose. The reported increase in total population dose from three in 10 years to one in 15 years is 4.82×10^{-2} person-rem per year, or 0.014 percent of the total population dose. These values are below the values associated with a small increase in population dose, as provided in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2. Thus, this increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

3. The reported increase in CCFP due to change in test frequency from one in 10 years to one in 15 years is 3.53×10^{-3} (0.353 percent). The cumulative increase in CCFP due to change in test frequency from three in 10 years to one in 15 years is 8.47×10^{-3} (0.847 percent). These values are considered small and are below the acceptance guidelines described in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2.

Based on the review of the risk assessment results, the NRC staff concludes that the increase in LERF is very small and consistent with the risk acceptance guidelines of RG 1.174. In addition, the increase in population dose (total integrated plant risk) and the change in the CCFP for the requested change are small and supportive of the LAR. 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 quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, the second condition in Section 4.2 of the SE for EPRI Report 1009325, Revision 2, is met.

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 Report No. 1009325, Revision 2, stipulates that in order to make the methodology in EPRI Report No. 1009325, Revision 2, acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a.

In Section 4.5.1 of the LAR, the licensee stated that the methodology in EPRI Report No. 1009325, Revision 2-A, incorporates the use of 100 L_a as the average leak rate for the pre-existing containment large leak rate accident case, and this value has been used in the plant-specific risk assessment. Accordingly, the third condition in Section 4.2 of the SE for EPRI Report No. 1009325, Revision 2, is met.

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

The fourth condition in Section 4.2 of the SE for EPRI Report No. 1009325, Revision 2, stipulates that in instances where containment over-pressure is relied upon for ECCS performance, a LAR is required to be submitted. In Attachment 1, Section 4.5.1, "Methodology," of the LAR, the licensee stated that WF3 does not rely on containment overpressure to assure adequate net positive suction head for ECCS performance. Accordingly, the fourth condition in Section 4.2 of the SE for EPRI Report No. 1009325, Revision 2, is not applicable

3.6 Summary

Based on the regulatory and technical evaluations above, the NRC staff finds that the licensee has adequately implemented its Containment Leakage Rate Testing Program consisting of ILRT and Local Leak Rate Testing (LLRT), CISI, and supplementary inspections. The results of the recent ILRTs, LLRTs and the CISI programs demonstrate acceptable performance of the WF3 containment and demonstrate 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, LLRT and CISI programs. The NRC staff finds that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting NEI 94-01, Revision 2-A, without undue risk to public health and safety. Therefore, the NRC staff concludes that it is acceptable to approve the proposed license amendment for WF3 to: (1) revise TS 6.15, "Containment Leakage Rate Testing Program," to adopt NEI 94-01, Revision 2-A and the Limitations and Conditions identified in the NRC staff's SE incorporated into NEI 94-01 Revision 2-A, as the implementation document, and (2) extend the current performance-based Type A test interval to 15 years on a permanent basis, provided acceptable performance history and other requirements discussed in this report are satisfied.

4.0 REGULATORY COMMITMENTS

The licensee made the following regulatory commitments in the August 28, 2014, LAR.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
WF3 will use the definition in Section 5.0 of NEI 94-01, Revision 2-A, for calculating the Type A leakage rate when future WF3 Type A tests are performed.		X	Following the NRC approval of this license amendment request.
The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provisions 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.		X	Following the NRC approval of this license amendment request.

In the SE dated June 25, 2008, the NRC staff concluded that the guidance in NEI 94-01, Revision 2-A, is acceptable for reference by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to the six limitations and conditions noted in Section 4.1 of the NRC SE for NEI 94-01, Revision 2. The first regulatory commitment stated by WF3, recited above, restates the first condition of the NRC SE for NEI 94-01, Revision 2. The second regulatory commitment stated by WF3, recited above, restates the fifth condition of the NRC SE for NEI 94-01, Revision 2. Accordingly, the regulatory commitments are acceptable, but redundant and, therefore, unnecessary.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment on August 12, 2015. The State official had no comments.

6.0 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 changes surveillance requirements. 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 published in the *Federal Register* on December 9, 2014 (79 FR 73106), and there has been no public comment on such finding. 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.

7.0 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.

Principal Contributors: M. Kichline
F. Farzam
S. Peng

Date: August 24, 2015

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

Sincerely,

/RA/

Michael D. Orenak, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures:
1. Amendment No. 244 to NPF-38
2. Safety Evaluation

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NAME	TLupold	SETurk w/noted changes	MKhanna	MOrenak	
DATE	07/08/2015	08/20/2015	08/24/2015	08/24/2015	

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