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

January 24, 2017

Site Vice President
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – ISSUANCE OF AMENDMENT RE: REVISION OF TECHNICAL SPECIFICATION 5.5.6 FOR EXTENSION OF TYPE A AND TYPE C LEAK RATE TEST FREQUENCIES (CAC NO. MF8305)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 312 to Renewed Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the technical specifications (TSs) in response to your application dated August 29, 2016, as supplemented by letter dated November 21, 2016.

The change to TS 5.5.6, "Primary Containment Leakage Rate Testing Program," would allow permanent extension of Type A and Type C leak rate test intervals through the adoption of Revision 3-A of Nuclear Energy Institute (NEI) 94-01 and the limitations and conditions specified in Revision 2-A of NEI 94-01 as the guidance documents for implementation of performance-based Option B of Appendix J to Title 10 of the *Code of Federal Regulations* Part 50, Option B, "Performance-Based Requirements." Based on the guidance in Revision 3-A of NEI 94-01, the change allows maximum interval for the Type A primary containment integrated leakage rate test to extend from once in 10 years to once in 15 years, and the Type C local leak rate test interval to extend to 75 months, provided acceptable performance history and other requirements are maintained.

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

Sincerely,

Diane L. Render, Ph.D.
Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures:
1. Amendment No. 312 to DPR-59
2. Safety Evaluation

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

ENTERGY NUCLEAR FITZPATRICK, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 312
Renewed License No. DPR-59

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated August 29, 2016, as supplemented by letter dated November 21, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-59 is hereby amended to read as follows:

2. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 312, are hereby incorporated into the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen S. Koenick, Acting Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: January 24, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 312

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
- 3 -	- 3 -

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
5.5-5	5.5-5

(4) ENO 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 without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus, components or tools.

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

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

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 2536 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 312, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

ENO shall implement and maintain in effect all provisions of the approved fire protections program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated November 20, 1972; the SER Supplement No. 1 dated February 1, 1973; the SER Supplement No. 2 dated October 4, 1974; the SER dated August 1, 1979; the SER Supplement dated October 3, 1980; the SER Supplement dated February 13, 1981; the NRC Letter dated February 24, 1981; Technical Specification Amendments 34 (dated January 31, 1978), 80 (dated May 22, 1984), 134 (dated July 19, 1989), 135 (dated September 5, 1989), 142 (dated October 23, 1989), 164 (dated August 10, 1990), 176 (dated January 16, 1992), 177 (dated February 10, 1992), 186 (dated February 19, 1993), 190 (dated June 29, 1993), 191 (dated July 7, 1993), 206 (dated February 28, 1994), and 214 (dated June 27, 1994); and NRC Exemptions and associated safety evaluations dated April 26, 1983, July 1, 1983, January 11, 1985, April 30, 1986, September 15, 1986, and September 10, 1992, subject to the following provision:

5.5.6 Primary Containment Leakage Rate Testing Program

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

- Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction, provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

a. The peak primary containment internal pressure for the design basis loss of coolant accident, P_a, is 45 psig.

b. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 1.5% of containment air weight per day.

c. The leakage rate acceptance criteria are:

1. Primary containment leakage rate acceptance criteria is $\leq$ 1.0 L_a. During plant 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 the Type A tests.

2. Air lock testing acceptance criteria are:

(a) Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq P_a$; and

(b) For each door seal, leakage rate is $\leq$ 120 scfd when tested at $\geq P_a$.

(continued)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 312

ENTERGY NUCLEAR FITZPATRICK, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-59

1.0 INTRODUCTION

By application dated August 29, 2016 (Reference 1), as supplemented by letter dated November 21, 2016 (Reference 2), Entergy Nuclear Operations, Inc. (Entergy, the licensee) submitted a license amendment request (LAR) to revise the James A. Fitzpatrick Nuclear Power Plant (JAFNPP) Technical Specification (TS) 5.5.6, "Primary Containment Leak Rate Testing Program." The supplement dated November 21, 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 or the Commission) staff's original proposed no significant hazards consideration determination.

The proposed change would allow for permanent extension of Type A and Type C leak rate test intervals through adoption of Revision 3-A of Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012 (Reference 3), and the limitations and conditions specified in Revision 2-A of NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008 (Reference 4), as the guidance documents for implementation of performance-based Option B of Appendix J to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Option B, "Performance-Based Requirements." This change would allow the maximum interval for the Type A primary containment integrated leakage rate test (ILRT) to extend from once in 10 years to once in 15 years, and the Type C local leak rate test (LLRT) interval to extend to 75 months, provided acceptable performance history and other requirements are maintained.

The licensee justified the proposed TS changes by providing historical plant-specific containment leakage testing program results and containment inservice inspection (ISI) program results and a supporting plant-specific risk assessment, consistent with the guidance in Revision 3-A of NEI 94-01 and the conditions and limitations contained in Revision 2-A of NEI 94-01.

Enclosure 2

2.0 REGULATORY EVALUATION

The following information explains the use of general design criteria (GDC) for JAFNPP. The construction permit for JAFNPP was issued by the Atomic Energy Commission (AEC) on May 20, 1970, and the operating license was issued on October 17, 1974. The plant design criteria for the construction phase is listed in the Updated Final Safety Analysis Report (UFSAR), Chapter 1.5, "Principal Design Criteria." The AEC published the final rule that added Appendix A to 10 CFR Part 50 in the *Federal Register* (36 FR 3255) on February 20, 1971, with the rule effective on May 21, 1971. In accordance with an NRC staff requirements memorandum, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (Reference 5), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971, which includes JAFNPP. However, the JAFNPP UFSAR, Chapter 16.6, "Conformance to AEC Design Criteria," evaluates JAFNPP against the 10 CFR 50, Appendix A, GDC. Also, the initial AEC safety evaluation (SE) of JAFNPP, dated November 20, 1972, Chapter 14.0, stated, "Based on our evaluation of the design and design criteria for the James A. FitzPatrick Nuclear Power Plant, we conclude that there is reasonable assurance that the intent of the General Design Criteria for Nuclear Power Plants, published in the Federal Register *(FR)* on May 21, 1971 as Appendix A to 10 CFR part 50, will be met." Therefore, the NRC staff reviews amendments to the JAFNPP license using the 10 CFR 50, Appendix A, GDC, unless there are specific criteria identified in the UFSAR.

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 Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J includes two options, Option A, "Prescriptive Requirements," and Option B, "Performance-Based Requirements," either of which may be chosen for meeting the requirements. JAFNPP was allowed a one-time extension of the then required 10-year maximum Type A test interval to 15 years by License Amendment No. 279, dated September 28, 2004 (Reference 6).

The testing requirements in Appendix J to 10 CFR Part 50 ensure that: (a) leakage through the containments or systems, and components penetrating the containments, does not exceed allowable leakage rates specified in the TSs and (b) the integrity of the containment structure is maintained during its service life. Option B has been voluntarily adopted and implemented by JAFNPP for meeting the requirements of Appendix J.

Option B specifies the performance-based requirements and criteria for preoperational and subsequent leakage rate testing. These requirements are met by performance of 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. Preoperational tests are required to be conducted at periodic intervals to ensure integrity of the overall containment system as a barrier to fission product release. Type A tests are based on the historical performance of the overall containment system and on the safety significance, and Type B and Type C tests are based on historical performance of each boundary and isolation valve. The leakage rate test results must not exceed the maximum allowable leakage rate (L_a) 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 system be conducted prior to each Type A test and at a periodic interval between tests, based on the performance of the containment system. These surfaces may affect the leaktight integrity of containment.

Section V.B.3 of Option B requires that the regulatory guide (RG) or other implementation document a licensee uses to develop a performance-based leakage testing program must be generally referenced in the TSs for the plant. 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 an RG.

The implementation document that is currently referenced in TS 5.5.6 is RG 1.163, "Performance-Based Containment Leak-Test Program," September 1995. Revision 0 of NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995, is endorsed by RG 1.163 as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B, subject to four regulatory positions delineated in Section C of RG 1.163. Revision 0 of NEI 94-01 includes provisions that allow the performance-based Type A test interval to be extended to up to 10 years, based upon two consecutive successful tests.

Section 50.55a of 10 CFR, "Codes and standards," contains the containment ISI program requirements that, in conjunction with the requirements of Appendix J, ensure the continued leaktight and structural integrity of the containment during its service life.

Section 50.65 of 10 CFR, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," paragraph (a), 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 50.36 of 10 CFR, "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 conditions for operation; (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

Revision 2-A of NEI 94-01 describes an approach for implementing the optional performance-based requirements of Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A test intervals to up to 15 years. In the NRC final SE for Revision 2 of NEI 94-01, dated June 25, 2008 (Reference 7), the NRC staff concluded that Revision 2-A of NEI 94-01 describes an acceptable approach for implementing the optional performance-based requirements of Option B and is acceptable for referencing by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to the specific limitations and conditions listed in Section 4.1 of the final SE for Revision 2 of NEI 94-01.

Revision 3-A of NEI 94-01 provides the JAFNPP operating license and guidance for extending Type C LLRT intervals beyond 60 months. This amendment also requests to extend the Type C test interval up to 75 months.

Revisions 2 and 3 of NEI 94-01 have been reviewed by the NRC staff and approved for use. The final SE for Revision 2 of NEI 94-01 documents the NRC staff's evaluation and acceptance of Revision 2 of NEI 94-01, subject to six specific limitations and conditions listed in Section 4.1 of the final SE dated June 25, 2008. The final SE for Revision 3 or NEI 94-01, dated June 8, 2012 (Reference 8), includes two specific limitations and conditions listed in Section 4.0 of that SE for the Type C test. Revision 3-A of NEI 94-01 and Revision 2-A of NEI 94-01 include their corresponding SEs.

Revision 4.0 of NUREG-1433, "Standard Technical Specifications – General Electric Plants (BWR/4)," incorporated Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-52, Revision 3, "Implement 10 CFR 50, Appendix J, Option B" (Reference 9), that provided guidance for specific changes to TSs for implementation of Option B.

3.0 TECHNICAL EVALUATION

3.1 Background

JAFNPP is a General Electric boiling-water reactor (BWR) with a Mark I water pool pressure suppression design containment. The primary containment consists of a carbon steel pressure vessel with a drywell composed of a cylindrical upper section and spherical lower section attached by large vent lines to a torus-shaped suppression chamber below, which is roughly halfway filled with water. The drywell houses the reactor vessel, recirculating loops, main steam lines, and other branch connections of the reactor coolant system. This arrangement limits primary containment pressurization from a design-basis loss-of-coolant accident (DBA LOCA) by channeling the steam and heated drywell atmosphere into the suppression pool, where condensation and cooling of the containment atmosphere occurs during the initial blowdown.

The primary containment provides the leaktight barrier against the potential uncontrolled release of fission products during a DBA LOCA. TS 5.5.6 identifies the primary containment allowable leakage rate (L_a) as 1.5 percent of the containment air weight per day at the calculated maximum DBA LOCA pressure (P_a) of 45 pounds per square inch gauge (psig). The design, fabrication, inspection, and testing of the primary containment vessel conform to the requirements for Class B vessels in the 1968 Edition of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) for nuclear vessels, including the 1968 Summer Addenda.

Section 9.2.3.1 of Revision 3-A of NEI 94-01 states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4 states that the assessment should be performed using the approach and methodology described in Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," August 2007 (Reference 10). The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.

In the final SE for Revision 2 of NEI 94-01, the NRC staff found the methodologies in Revision 2-A of NEI 94-01, and EPRI Final TR-1009325, Revision 2, are acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied. The NRC staff set forth the following conditions related to referencing the methodology in EPRI Final TR-1009325, Revision 2:

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

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 of the final SE for Revision 2 of NEI 94-01.

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

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

3.2 Licensee's Proposed Changes

JAFNPP TS 5.5.6 currently states, in part:

> This program implements 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, as modified by the following exceptions:
>
> - NEI 94 -01-1995. Section 9.2.3: The first Type A test performed after the March 7, 1995 Type A test shall be performed no later than March 7, 2010.
>
> - Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction, provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the program.

Entergy's proposed amendment would revise JAFNPP's TS 5.5.6 to state:

> This program implements the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exception:
>
> - Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction, provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

With this change, JAFNPP will implement Revision 3-A of NEI 94-01, and the limitations and conditions of Section 4.1 of the final SE for Revision 2-A of NEI 94-01. Revision 3-A of NEI 94-01 provides that extension of the Type A test interval to 15 years be based on two consecutive successful Type A tests (performance history) and other requirements stated in Section 9.2.3. 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 is 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 significant leakage potential were to develop.

Revision 0 of NEI 94-01 provided that a maximum interval of 120 months could be allowed for Type B tests and Revision 3-A of NEI 94-01, now also provides that a maximum interval of 75 months could be allowed for Type C tests, with a limited provision for extension or grace period up to 9 months allowed for these LLRTs.

The existing JAFNPP TS 5.5.6 exception to NEI 94-01 regarding next Type A test performance by a specified a date is no longer needed and may be removed, as that date has passed with the test having been performed.

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

3.2.1 Previous Type A ILRT Results

Entergy stated in its LAR dated August 29, 2016, that the last Type A ILRT was completed in October 2008. Previous Type A testing confirmed that the containment structure leakage is acceptable, with considerable margin with respect to the TS acceptance criterion of 0.75 L_a, at pressure P_a, as shown in Table 1 below:

Table 1

Type A - Integrated Leakage Rate Testing History

Test Date	As Found Test Result (% Weight per day)	As Found Test Result (% Weight per day)	Acceptance Criteria (L_a) (% Weight per day)
June – 1990	0.4035 [1]	0.2704	0.5
March – 1995	0.31999 [1,2]	0.06299	0.5
October - 2008	0.8052 [1,3]	0.6045	1.125

The current performance criterion is L_a, which is 1.125 percent by weight of the containment air per day at the peak DBA LOCA pressure of 45 psig. The LAR-provided ILRT results show substantial margin has been maintained relative to the performance criterion, and specifically for the last two ILRTs, so an extended interval would be allowed by program guidance for JAFNPP. In addition, no adverse trend is apparent in the ILRT results that would suggest the performance criterion might be exceeded with the requested extension of the maximum ILRT interval to 15 years.

3.2.2 ISI Program

Note 1: All Type A testing has been performed at test pressures greater than P_a.

Note 2: Data analysis technique used – absolute method, leakage rates were calculated using total time leakage per American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-1987.

Note 3: Data analysis technique used – absolute method and the leakage rates were calculated using the total time analysis equations as described in BN-TOP-1.

The Type A test acceptance criteria is as follows, in accordance with TS 5.5.6:

a. The peak primary containment internal pressure for the design basis loss of coolant accident, P_a, is 45 psig.

b. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 1.5% of containment air weight per day.

c. The leakage rate acceptance criteria are:

1. Primary containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During plant 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 the Type A tests.

The containment concrete is used only to meet intended functions of shielding and structural support. The containment concrete does not serve as a pressure retaining function and, therefore, IWL is not applicable to a BWR Mark I containment.

a. ISI Program for Containment Metal Liner – Subsection IWE

In the LAR, the licensee stated that the second 10-year Interval for IWE containment inspections at JAFNPP commenced on March 1, 2007, coincident with the start of the fourth 10-year ISI program interval.

JAFNPP is committed to the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," the applicable requirements of IWE (requirements for Class MC and metal liners of Class CC components) of 2001 Edition through the 2003 Addenda. The licensee also stated that a general visual inspection of the accessible surfaces of the drywell (interior of the primary containment) is performed to examine coatings, the moisture barrier at the concrete-to-drywell shell plates interface, and an overall condition of the structural integrity of the interior primary containment steel and concrete surfaces. Minor discoloration and small areas of coating damage from apparent activities performed in the past were observed. These imperfections do not affect the integrity of the containment and are considered acceptable. All bolted connections were observed to be intact and free of unacceptable discontinuities. This examination included the containment dome. Engineering evaluations were not required from the general visual inspections performed during Refueling Outage (RO) 15.

The JAFNPP ISI schedule is indicated in Table 2 below:

Table 2

JAFNPP Containment Inservice Inspection Periods

Inspection Period	Period Start Date	Period End Date
1st Interval Period 1	September 28, 1997	March 27, 2001 (RO13 and RO14)
1st Interval Period 2	March 28, 2001	November 30, 2004 (RO15 and RO16)
1st Interval Period 3	December 1, 2004	February 28, 2007 (RO17)
2nd Interval Period 1	March 1, 2007	December 31, 2010 (RO18 and RO19)
2nd Interval Period 2	January 1, 2011	December 31, 2013 (RO20)
2nd Interval Period 3	January 1, 2014	December 31, 2016 (RO21 and RO22)
3rd Interval Period 1	January 1, 2017	December 31, 2019 (RO23)
3rd Interval Period 2	January 1, 2020	December 31, 2023 (RO24 and RO25)
3rd Interval Period 3	January 1, 2024	December 31, 2026 (RO26 and RO27)

Note: If the second and/or third periods require an adjustment, IWA-2430 allows the inspection interval to be reduced or extended by as much as 1 year to enable an inspection to coincide with a plant outage.

Note: The dates for the third interval are proposed dates. The third interval plan has not been issued at this time.

Entergy stated in the LAR that in accordance with IWE-1231, a minimum of 80 percent of the surface area defined in Examination Category E-A in Table IWE-2500-1, and areas defined in IWE-1240, are required to remain accessible for inspection. Entergy also stated that IWE-1230 defines the minimum requirements for accessibility for examination from at least one side of the vessel. The total surface area that is inaccessible amounts to an area of 1,829.1 square feet (sq. ft.) (inside surface examination). The total interior surface of the primary containment (with the torus drained) amounts to approximately 64,500 sq. ft. as per Chicago Bridge and Iron Company Document, Drywell and Suppression Chamber Cleaning and Painting Instructions. Therefore, approximately 97 percent of the total inside surface area is accessible for examination.

Entergy further stated that for Class MC applications, JAFNPP shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, JAFNPP shall provide the following in the Owners Activity Report-1, as required by 10 CFR 50.55a(b)(2)(ix)(A):

- A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
- An evaluation of each area, and the result of the evaluation; and
- A description of necessary corrective actions.

Entergy actively participates in various nuclear utility owners groups and ASME Code committees to maintain cognizance of ongoing developments within the nuclear industry. Industry operating experience is also continuously reviewed to determine its applicability to JAFNPP. Adjustments to inspection plans and availability of new, commercially available technologies for the examination of the inaccessible areas of the containment would be explored and considered as part of these activities.

In the LAR, the licensee also stated that the implementation of the proposed change for TS 5.5.6 will be revised by replacing the reference to RG 1.163 with reference to Revision 3-A of NEI 94-01. This will require that a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leaktight integrity be conducted. This inspection 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 for the Type A test has been extended to 15 years in accordance with the following sections of Revision 3-A of NEI 94-01:

- Section 9.2.1, "Pretest Inspection and Test Methodology"
- Section 9.2.3.2, "Supplemental Inspection Requirements"

In addition to the IWE examinations scheduled in accordance with the containment ISI program, the performance of inspections utilizing Surveillance Test (ST)-15B, "Suppression Chamber and Drywell Deterioration Inspection," will be utilized to ensure compliance with the visual inspection requirements of TS SR 3.6.1.1.1 and Revision 3-A of NEI 94-01. The inspections utilizing ST-15B are performed each refueling outage.

b. Primary Containment Coatings Program

In the LAR, the licensee stated that all coatings used inside primary containment are qualified for JAFNPP-specific normal and DBA environmental conditions in accordance with ANSI N5.12-i 974 and ANSI N101.2-1972.

The following procedures are among the containment coating controls identified in response to NRC Generic Letter 98-04:

- ST-15B, Suppression Chamber and Drywell Deterioration Inspection
- IS-M-01, Preparation and Painting of Plant Structures, Components and Concrete Items

Coatings assessments are completed each refueling outage in accordance with ST-15B, and work orders are initiated for any area having coating failures (peeling, flaking, blistering, cracking, or mechanically-induced failure such as scraping). Coatings assessments have also been performed in accordance with the drywell preservation and torus preservation programs and concluded that both the drywell and the torus are in good condition. The torus and drywell are visually inspected each outage. To date, no evidence of significant coatings degradation has been identified.

3.2.3 Type B and Type C LLRT Program

The Type B and Type C testing program at JAFNPP requires testing of electrical penetrations, airlocks, hatches, flanges, and containment isolation valves in accordance with Option B of 10 CFR Part 50 and RG 1.163. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Type B and Type C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. In accordance with TS 5.5.6, the allowable maximum pathway total Type B and Type C leakage is 0.6 L_a, where L_a equals 320 standard liters per minute (SLM). In addition to the TS Limit, an outage performance limit of 160 SLM, or 0.5 L_a is imposed by the program, as shown in the Table 3 below:

Table 3

JAFNPP Type B and Type C LLRT Combined As-Found (AF) and As-Left (AL) Trend Summary

RO	**RO16 (2004)**	**RO17 (2006)**	**RO18 (2008)**	**RO19 (2010)**	**RO20 (2012)**	**RO21 (2014)**
AF Min Path (SLM)	99.52	43.15	77.642	65.21	298.104	160.958
Fraction of L_a	0.311	0.135	0.243	0.204	0.9315	0.503
AL Max Path (SLM)	45.859	72.393	101.118	74.958	137.481	131.145
Fraction of L_a	0.143	0.226	0.316	0.234	0.430	0.410
AL Min Path (SLM)	34.722	45.399	66.344	50.984	74.88	80.713
Fraction of L_a	0.1085	0.142	0.207	0.159	0.234	0.252

Based on the information provided in Table 3 above, the NRC staff also determined that the licensee's ILRT, LLRT, and containment examination programs to periodically examine, monitor, and manage age-related and environmental degradation of JAFNPP's containment support extending the Type A ILRT test out to a maximum of 15 years.

JAFNPP TS 5.5.6 acceptance criterion for combined Type B and Type C test total is 0.6 L_a. As detailed in NEI 94-01, this acceptance criterion is evaluated minimum pathway for as-found (AF) values and maximum pathway for as-left (AL) values. The AF minimum pathway total provides an assessment of the leakage testing and the corrective action program's effectiveness for ensuring penetration leakage potential is kept low throughout each operating cycle such that sufficient margin to L_a is maintained to accommodate some increase in non-penetration leakage potential between ILRTs. The AL maximum pathway total criterion is permissive for restoring primary containment operability and requires margin to accommodate increases in leakage potential between outages where leakage testing is performed, as well as a failure of a containment isolation valve to close. In its supplemental letter dated November 21, 2016, the licensee indicated that the 2012 combined Type B and Type C AF test total did not exceed any acceptance criteria and characterized the statements in ANSI 56.8-1994 and NEI 94-01 (1995) regarding acceptance criteria for AF testing as suggestions from supporting documents. However, as the discussion in *Federal Register* notice 60 FR 49495, dated September 26, 1995, indicated, in response to a question regarding regulation and implementation documents being developed in such a manner that they could be objectively and consistently inspected and enforced:

> The NRC wishes to retain the current practice which requires its review and approval of changes to Appendix J performance limits and surveillance requirements. Therefore, the NRC has required that the regulatory guide should be specified in the technical specifications, ...the review and endorsement of a industry guideline in a regulatory guide, and the general reference of the regulatory guide in plant technical specifications, will provide a common understanding on the measures of compliance.

Appendix J of 10 CFR 50 states, in part:

> Section III, "Performance-Based Leakage-Test Requirements"
>
> *B. Type B and C Tests*, "The tests must demonstrate that the sum of the leakage rates at accident pressure of Type B tests, and pathway leakage rates from Type C tests, is less than the performance criterion (L_a) with margin, as specified in the Technical Specification."
>
> Section V, "Application"
>
> *B. Implementation*
>
> 3. The regulatory guide or other implementation document used by a licensee or applicant for an operating license under this part or a combined license under part 52 of this chapter to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification 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.

In the voluntary implementation of Option B, NRC objectives include limiting the prescriptive or detailed content of both the regulation provision and the associated plant TS changes and have most details left to the guidance documents identified in the TSs. In justifying the longer ILRT intervals with less frequent assessment of primary containment leakage potential, more reliance was placed on LLRT where most potential leakage is expected to occur and on containment inspection programs to identify and allow remediation of developing conditions with the primary containment boundary not detectable by LLRT before they could develop significant leakage or structural impairment potential. In considering the industry-proposed change to allow ILRT intervals to routinely extend to a maximum of 15 years, the reliance on LLRT monitoring was reemphasized. Revision 0 of NEI 94-01 states the following:

> ANSI/ANS 56.8-1994 also specifies surveillance acceptance criteria for Type B and Type C tests. The ANSI/ANS 56.8-1994 definition is that the combined leakage rate for all penetrations subject to Type B or Type C tests is limited to less than or equal to 0.6 L_a, when determined on a MNPLR [minimum pathway leakage rate] basis from As-found LLRT results; and limited to less than or equal to 0.6 La, as determined on a Maximum Pathway Leakage Rate (MXPLR) basis from the As-left LLRT results.
>
> Due to the performance-based nature of Option B to Appendix J and this guideline, it is recommended that acceptance criteria for the combined As-found leakage rate for all penetrations subject to Type B or Type C testing be the same as that defined in ANSI/ANS 56.8-1994 with the following additions. The combined As-left leakage rates determined on a MXPLR basis for all penetrations shall be verified to be less than 0.6 La prior to entering a mode where containment integrity is required following an outage or shutdown that included Type B and Type C testing only. The combined As-found leakage rates determined on a MNPLR basis for all penetrations shall be less than 0.6La at all times when containment integrity is required. These combined leakage rate determinations shall be done with the latest leakage rate test data available, and shall be kept as a running summation of the leakage rates.

The corresponding verbiage in Revision 3-A of NEI 94-01 and Revision 2-A of NEI 94-01 now states as follows:

> ANSI/ANS-56.8-2002, Section 6.4.4 also specifies surveillance acceptance criteria for Type B and Type C tests, and states that the combined (as-found) leakage rate of all Type B and Type C tests shall be less than 0.6 L_a when evaluated on a MNPLR basis at all times when containment operability is required. Moreover, the combined leakage rate for all penetrations subject to Type B and Type C tests shall be less than or equal to 0.6 L_a as determined on an MXPLR basis from the as-left LLRT results. ... These combined leakage rate determinations shall be done with the latest leakage rate test data available, and shall be kept as a running summation of the leakage rates.

The NRC staff's SEs incorporated into Revision 3-A of NEI 94-01 and Revision 2-A of NEI 94-01 contain a section entitled, "The Impact of the Proposed Change Should be Monitored Using Performance Measurements Strategies," which states:

> In addition to maintaining the defense-in-depth philosophy as described in Section 3.2.2 of this SE, the applicants for TS amendments will continue to perform containment inspections during the Type A test interval as discussed in Sections 3.1.3 and 3.1.4 of this SE.
>
> As documented in NUREG-1493, industry experience has shown that most ILRT failures result from leakage that is detectable by local leakage rate testing (Type B and C testing). Specific testing frequencies for the local leak rate tests are reviewed prior to every refueling outage (18-month cycle). An outage scope document is issued to document the local leak rate test periodically and to ensure that all pre-maintenance and post-maintenance testing is complete. The post-outage report provides a written record of the extended testing interval changes and the reasons for the changes based on testing results and maintenance history. Based on the above measures, the LLRT program will provide continuing assurance that the most likely sources of leakage will be identified and repaired.
>
> ANSI/ANS-56.8-2002, Section 6.4.4, also specifies surveillance acceptance criteria for Type B and Type C tests and states that: "The combined [as-found] leakage rate of all Type B and Type C tests shall be less than 0.6 L_a when evaluated on a minimum pathway leakage rate basis, at all times when containment operability is required." It states, moreover, that: "The combined leakage rate for all penetrations subject to Type B and Type C test shall be less than or equal to 0.6 L_a as determined on an maximum pathway leakage rate basis from the as-left LLRT results." These combined leakage rate determinations shall be done with the latest leakage rate test data available, and shall be kept as a running summation of the leakage rates.
>
> The containment components' monitoring and maintenance activities will be conducted according to the requirements of 10 CFR 50, Appendix J, and 10 CFR 50.55a.
>
> The above provisions are considered to be acceptable performance monitoring strategies for assuring that the risk of the proposed change will remain small.

In its November 21, 2016, letter, the licensee indicated that a corrective action report (CR-JAF-2012-07378) had been initiated regarding the 2012 refueling outage combined Type B and Type C LLRT minimum pathway leakage total. Although the history of Type B and Type C testing combined totals do not show consistent satisfactory performance (AF minimum pathway), the LAR included Tables 3.4.5-2 and 3.4.5-3 for valves whose testing results during RO20 in 2012 and RO21 in 2014 exceeded their individual administrative limits. These valves would also appear to have significantly contributed to the combined totals. Causes that could be determined were noted, as were corrective actions or evaluations. The combined AF minimum pathway total in the 2014 refueling outage was acceptable but above typical historical

values. The leakage rate testing program assigns individual penetrations/valves an administrative limit, and corrective actions are determined for that penetration/valve accordingly. The corrective action and maintenance rule monitoring programs are expected to highlight and drive additional actions to prevent recurrence. Tables 3.4.5-2 and 3.4.5-3 in the LAR show one component, valve 23M0V-15, to have exceeded its administrative limit moderately in both outages, and the measured leakage did not change significantly from 2012 to 2014.

The data in Table 3.4.5-1 of the LAR show that LLRT totals were generally acceptable and supportive of the requested extension of ILRT maximum interval to 15 years, and Type C testing maximum intervals to 75 months, for qualifying valves.

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

In the NRC final SE dated June 25, 2008 (Reference 7), the staff concluded that the guidance in Revision 2-A of NEI 94-01 is acceptable for reference by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to six conditions. The requirements of NEI 94-01 remained essentially the same from the original version through Revision 2, with the more significant changes being that the regulatory positions of RG 1.163 were incorporated, and the maximum ILRT interval was 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 Revision 3 of NEI 94-01, and were not carried forward into the NRC final SE for Revision 3. However, the licensee explicitly identified the limitations and conditions specified in Revision 2-A of NEI 94-01 in the proposed TS.

a. NRC Condition 1

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

The licensee stated in the LAR that it would be using the definition in Revision 2-A of NEI 94-01, Section 5.0. The requirement is the same in Revision 3-A, Section 5.0. Therefore, the licensee addressed and satisfied NRC Condition 1.

b. NRC Condition 2

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

The licensee provided the current schedule of containment inspections in LAR Table 3.4.3-1 and described the general visual inspection performed each refueling outage in LAR section 3.4.1. Therefore, the licensee addressed and satisfied NRC Condition 2.

c. NRC Condition 3

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

The licensee described in Sections 3.4.3 and 3.5 of the LAR the inaccessible Class MC areas and the augmented examinations or potential for augmented examinations. Therefore, the licensee addressed and satisfied NRC Condition 3.

d. NRC Condition 4

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

This condition is intended to verify any major modification or maintenance repair of the primary since the last ILRT has been appropriately accompanied by a either a structural integrity test or ILRT and that any plans for such major modification also include appropriate pressure testing. The licensee indicated in Section 3.5.1 of the LAR that in 2005, after repair of cracked torus shell due to the high pressure coolant injection pump exhaust line discharge cyclic stress fatigue, the primary containment had been pressurized to the peak accident pressure of 45 psig, and accessible areas of the containment were subjected to a visual examination. An ILRT was performed in 2008, and the LAR stated that there are no major modifications planned that would require either a structural integrity test or ILRT. Therefore, the licensee addressed and satisfied NRC Condition 4.

e. NRC Condition 5

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

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

f. NRC Condition 6

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

This condition is not applicable to JAFNPP, as it was not licensed under 10 CFR Part 52.

3.4 NRC Conditions in Revision 3-A of NEI 94-01

In the final SE for Revision 2 of NEI 94-01, the NRC staff concluded that the guidance in Reference 3 is acceptable for reference by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to two conditions.

a. NRC Condition 1

> NEI TR 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The 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 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.

b. NRC Condition 2

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

The licensee indicated in the LAR that the JAFNPP post-outage reports will include the margin between the Type B and Type C minimum pathway leak rate summation value adjusted for understatement and the acceptance criterion. Should the Type B and Type C combined totals exceed an administrative limit of 0.5 L_a but be less than the TS acceptance value (performance criterion) of 0.6 L_a, an analysis will be performed, and a corrective action plan prepared, to restore and maintain the leakage summation margin to less than the administrative limit. The LAR also stated that JAFNPP will apply the 9-month grace period only to eligible Type C tested components and only for non-routine emergent conditions. The licensee acknowledges these two conditions and the likelihood that longer test intervals would increase the understatement of actual leakage potential, given the method by which the totals are calculated, and will assign additional margin for monitoring acceptability of results by administrative limits and understatement

contribution adjustments. Therefore, the licensee addressed and satisfied NRC Conditions 1 and 2 of Revision 3-A of NEI 94-01.

3.5 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval to once in 15 years. The risk assessment was provided in Attachment 4 to the LAR. Additional information was provided by the licensee in response to NRC requests for additional information (RAI) in Reference 2.

In Section 1.0 of Attachment 4 to the LAR, the licensee stated that the plant-specific risk assessment follows the guidelines stated in Reference 3; the methodology used in of EPRI TR-104285; Revision 4, of NEI's guidance, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals"; the regulatory guidance on the use of PRA as stated in Revision 2 of RG 1.200, as applied to ILRT interval extensions; risk insights in support of a request for a plant's licensing basis, as outlined in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"; and the methodology used in EPRI TR-1009325, Revision 2-A. Additionally, the licensee used the methodology from Calvert Cliffs Nuclear Power Plant to assess the risk from undetected containment leaks due to steel liner corrosion.

Section 2.0 of Attachment 4 to the LAR states that the basis for the current 10-year test interval is provided in Section 11.0 in Revision 0 of NEI 94-01. In response to APLA RAI-2, the licensee also provided the basis for extending the ILRT to a 15-year test interval per Reference 3. The licensee clarified that Reference 3 supports using EPRI TR-1009325, Revision 2-A, for performing risk impact assessments in support of ILRT extensions. The guidance provided in Appendix H of EPRI TR-1009325, Revision 2-A, builds on the EPRI risk assessment methodology, EPRI TR-104285. This methodology was followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

The licensee addressed each of the four conditions listed in Section 3.1 of this SE. A summary of how each condition has been met is provided in the sections below.

3.5.1 Technical Adequacy of the PRA

The first condition of the final SE for Revision 2 of NEI 94-01, 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.

Consistent with the information provided in NRC Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation" (Reference 11), the NRC staff will use Revision 2 of RG 1.200 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 Revision 2 of NEI 94-01, and EPRI TR-1009325, Revision 2, the NRC staff states that Capability Category (CC) I of the ASME PRA standard shall be applied as the standard for assessing PRA quality for IRLT extension applications, since approximate values of core damage frequency (CDF) and large early release

frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

The JAFNPP PRA model used for the ILRT application is a Level 1 and Level 2 model for internal events, including internal flood, for at-power operations. The licensee employs a multifaceted approach to establishing and maintaining the technical adequacy of the PRA models. This approach includes procedures for maintaining and updating the PRA model and utilizing self-assessments and independent peer reviews. Following each periodic PRA model update, the licensing performs a self-assessment to assure that the PRA quality and expectations for all current applications are met. However, the JAFNPP model used for this application is referred to as the following:

> .....outside of the internally required four year periodic update cycle of the model of record, JAF-NE-09-00001 "Fitzpatrick Probabilistic Safety Assessment (PSA), Rev 4", (Reference 45). The results are considered reasonable for the ILRT extension risk analysis because potential changes to the model are assessed using the process described in this section, and because plant specific initiating events, failure rates, and maintenance unavailability changes are not expected to result in an increase in the relatively low JAF PRA total CDF, LERF, and corresponding calculated delta LERF sufficient to challenge the conclusion of the analysis with respect to the risk acceptance criteria in Regulatory Guide 1.174.

The licensee stated that in September 2009, the Boiling Water Reactor Owner's Group performed a full-scope peer review of all JAFNPP technical elements in the internal events at-power PRA, using NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard"; the ASME/ANS RA-Sa 2009 standard; and Revision 2 of RG 1.200:

- Initiating Events Analysis (IE)
- Accident Sequence Analysis (AS)
- Success Criteria (SC)
- Systems Analysis (SY)
- Human Reliability Analysis (HR)
- Data Analysis (DA)
- Internal Flooding (IF)
- Quantification (QU)
- LERF Analysis (LE)
- Maintenance and Update Process (MU)

During the JAFNPP PRA model peer review, the technical elements identified above were assessed with respect to CC II criteria. The peer review team identified 21 finding-level facts and observations (F&Os) that did not meet the CC II criteria. For these F&Os, 11 were open, and they were summarized and evaluated for this ILRT application in Table A-1 of Attachment 4 to the LAR. Regarding the 11 open F&Os, the staff requested clarifications associated with SRs HR-G1 (APLA RAI-10a), QU-A3 (APLA RAI-10b), LE-C4 (APLA RAI-10c), and IFSN-A6 (APLA RAI-10d) to determine whether the impact of these findings was appropriately considered in estimating the risk associated with extending the ILRT test interval. The remaining 10 of the

21 F&Os were dispositioned and were summarized in Table A-2 of Attachment 4 to the LAR. The NRC staff reviewed the 10 F&Os that were dispositioned by the licensee and concluded that they were evaluated and addressed appropriately for this application.

The following summarizes the staff's requests and the licensee's response to APLA RAI-10a to APLA RAI-10d:

APLA RAI-10a: Regarding HR-G1, the staff was concerned that very low joint human error probability (HEP) values would lead to a non-conservative risk evaluation result for this application. The staff requested the licensee to provide any joint HEP floor value applied to the cutsets that contain multiple HEPs for the re-analysis of the combined HEPs. If the floor value is significantly less than the value stated in NUREG-1792 or EPRI TR-1021081, provide justification. If there is no floor value applied to those cutsets that contain multiple HEPs, provide justification and a sensitivity evaluation using a joint HEP floor value of greater than 1E-6 to show that these low joint HEP values have no significant impact on the overall results stated in the LAR.

In response to APLA RAI-10a, the licensee provided two sensitivity studies by using joint HEP values of 1E-5 and 1E-6, respectively. While there were substantial increases in some of the various risk metrics (e.g., roughly factors of 3 for increase in total LERF, total dose rate, total increase in dose rate and ΔLERF using the higher floor value of 1E-5), these studies demonstrated that using a minimum joint HEP floor does not affect the conclusions of the ILRT extension risk analysis because the risk metric thresholds remained non-exceeded.

APLA RAI-10b: Regarding QU-A3, the staff requested the licensee to provide the quantitative results from the sensitivity evaluation performed at the time of the Revision 2 of RG 1.200 peer review. The results should include the effect on both CDF and LERF due to parametric/state-of-knowledge uncertainty, since the licensee did not include LERF for this F&O.

In response to APLA RAI-10b, the licensee performed a new sensitivity evaluation by redefining the type codes based solely on equipment type and failure mode to allow a state-of-knowledge correlation. When multiple failure rates exist for different systems within a type code, a bounding failure rate is used for the sensitivity evaluation. The parametric uncertainty was computed using a Monte Carlo method with 10,000 samples. This large number of samples ensured stable results with robust estimates of key percentiles. The total CDF increases from 2.40E-6 to 2.72E-6/year (yr), and the total LERF increases from 2.69E-7 to 2.75E-7/yr. These minor increases did not change the conclusion of the LAR. (Note that these total CDF and LERF values are extracted from page 67 of 69 of Attachment 2 to Reference 2. There are typographical errors in the total CDF and LERF values listed.)

APLA RAI-10c: Regarding LE-C4, the licensee stated that there is no impact resulting from using point estimates in the Level 2 model for success branches, since the system-related success probabilities are approximately equal to 1. The staff requested the licensee to provide all success probabilities to show that their values have minimal impact on the quantification for this application.

In response to APLA RAI-10c, the licensee confirmed that most success probabilities are greater than 0.9 and are sufficiently close to 1, except there are three probabilities that are not greater than 0.9. However, for the success probabilities that are less than 0.9, quantifying only the system portion of the failure logic gate and taking the complement of the system failure probability shows system success to be 0.999, which is approximately equal to 1.0. Therefore, the total LERF used in the ILRT application is not impacted, and there is no impact on the change in risk estimate of the ILRT extension analysis.

APLA RAI-10d: Regarding IFSN-A6, the licensee did not include an assessment to meet CC II. The staff requested the licensee to provide the results from the analysis related to IFSN-A6 to show that CC II requirements are met, or explain how meeting only CC I requirements does not have impact on the final conclusions of this application.

In response to APLA RAI-10d, the licensee stated that CC I is sufficient to provide an estimate of the total internal flooding CDF and LERF as used in the ILRT extension PRA analysis. Meeting CC II could provide additional qualitative discussion regarding the impact of flood-induced mechanisms that are not formally addressed using the mechanisms in CC III, but would not impact the total CDF and LERF used in the ILRT extension analysis.

Based on the above, the staff concluded that the internal events PRA was reviewed against the applicable SRs in ASME/ANS-RA-Sa-2009 as endorsed and clarified by Revision 2 of RG 1.200.

In Section 3.2.4.2 of the final SE for Revision 2 of NEI 94-01, the NRC staff states that:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals." This section also states that: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [(i.e., EPRI Report No. 1009325, Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed." This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

The licensee evaluated the impact of external events in Section 3.3.2 of Attachment 1 to the LAR. The licensee's assessment included an estimate of internal fires, seismic, high winds, floods, and other hazards, based on the individual plant examination of external events (IPEEE).

The fire risk was calculated during the IPEEE using the EPRI fire-induced vulnerability evaluation (FIVE) method. The IPEEE was published more than 20 years ago, and the licensee did not update its fire risk model for this application. Therefore, the staff requested the licensee to either identify any gaps between the JAFNPP IPEEE fire risk model and the "internal fire technical elements" required by Revision 2 of RG 1.200, or demonstrate that addressing the

requirements would have no impact on the final results in the LAR by providing a sensitivity evaluation to show that the use of the IPEEE results remains conservative by updating the IPEEE results, including:

1. Reanalysis of all containment bypass scenarios leading to LERF from the internal events PRA, replacing any random failure probabilities with appropriate fire-induced values (e.g., NUREG/CR-7150 values for motor-operated valves for interfacing-systems loss-of-coolant accident (ISLOCAs)). This should include any ISLOCA pathways previously screened out due to low random failure probability, as the probability could be significantly higher if fire is induced.

2. Reanalysis of the IPEEE fire LERF using updated fire frequencies and, if suppression was credited, updated non-suppression probabilities from NUREG-2169.

In response to this RAI, the licensee performed a sensitivity analysis in accordance with the above criteria. All IPEEE fire scenarios were revised to include the updated ignition frequencies and non-suppression probabilities. An assessment of which scenarios would include risk contribution from the bypass scenarios not previously analyzed in the IPEEE was performed, which included the following methodology:

1. Conservatively consider all fire scenarios as impacting the valves whose failure would potentially lead to a containment bypass

2. Review the risk-significant fire scenarios, considering the cable routing information available in the JAFNPP safe shutdown analysis.

3. Remove the fire-induced failure of valves whose failure would potentially lead to a containment bypass scenario from the fire scenarios in fire areas where no cables associated with the valve are routed.

The reanalyzed fire LERF is 6.20E-6, an increase of approximate 186 percent of the fire LERF result stated in the LAR. However, this value is still within the risk acceptance guidelines stated in RG 1.174.

The external events risk assessment provided in the LAR provided no quantification of seismic risk. Section 3.3.2 of Attachment 1 to the LAR states that in the JAFNPP IPEEE, a seismic margin assessment (SMA) was performed. An SMA is a deterministic and conservative evaluation that does not calculate risk on a probabilistic basis. In response to APLA RAI-1b, the licensee clarified that an estimate from Generic Issue (GI) 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment" (Reference 12), based on the U.S. Geological Survey seismic hazard curves, was used for quantification (bounding CDF = 6.1 E-6 per year from "Weakest Link Model" stated in GI-199. The licensee indicated that the conclusions stated in GI-199 were used for estimating seismic CDF in this application.

In addition to internal fires and seismic events, the licensee performed a plant hazard and design information review for high winds, floods, and other hazards in the IPEEE, for conformance with the Standard Review Plan criteria. The licensee performed an analysis in the

IPEEE to show that the external hazard frequencies of those hazards that could not be screened out were acceptably low and have negligible impact on this application.

The licensee's updated estimates of change in LERF due to ILRT extension for combined internal and external events is discussed in Section 3.5.2 of this SE. The staff finds the licensee's analysis of the impact of external events acceptable for the ILRT application.

In summary, the licensee has evaluated its internal events PRA against the ASME/ANS RA-Sa-2009 PRA standard and Revision 2 of RG 1.200, addressed or evaluated the impact of the findings developed during the peer review of its internal events PRA for applicability to the ILRT interval extension, and included a quantitative assessment of the contribution of external events. Additionally, the staff concluded that the impact from external events is within the acceptance guidelines stated in RG 1.174. Based on the above, the NRC staff concluded that the PRA used by the licensee is of sufficient technical adequacy to support the evaluation of changes to ILRT frequency. Accordingly, the first condition is met.

3.5.2 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 is consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.5 of the final SE for Revision 2 of NEI 94-01. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-roentgen equivalent man (rem) per year, or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points. Additionally, for plants that rely on containment overpressure for net positive suction (NPSH) for ECCS injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. The associated risk metrics include LERF, population dose, and CCFP, but not CDF, because JAFNPP does not rely on containment overpressure for ECCS performance. This is further discussed in Section 3.5.4 of this SE.

The licensee provided a plant-specific risk assessment in Attachment 4 to the LAR and a revision of this assessment in Attachment 2 to Enclosure 1 of the licensee's response to the RAIs in its letter dated November 21, 2016. 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:

1. The reported increase in LERF for internal events is 2.17E-08/yr; this value increases slightly to 2.20E-08/yr if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included. As reported in response to APLA RAIs 1a, 3, and 6, the increase in LERF for combined internal and external events is 3.16E-07/yr for both units. The risk contribution from external events includes the effects of internal fires, seismic events and high winds, floods, and other external hazards as discussed in Section 3.2.1 of this SE. These

changes in risk are 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. In response to APLA RAIs 1a, 3, and 6, the licensee estimated the total LERF for internal and external events as 8.29E-06/yr. The total LERF, given the increase in ILRT interval, is below the acceptance guideline of 1E-05/yr in RG 1.174 for a "small" change.

2. The reported change in Type A ILRT frequency from three in 10 years to once in 15 years results in a reported increase in the total population dose of 0.0087 person-rem/yr. In response to APLA RAI-4, the licensee clarified the use of scaling the population dose from the Peach Bottom Atomic Power Station produced a more conservative result. The calculations of the population dose were verified by the licensee in response to APLA RAIs 7 and 9. Table 5-13 in the original LAR provided the total dose increase, which included the increases in Classes 3a and 3b, and the decrease in Class 1, but Table 5-21 only provided the dose increase of Class 3b. Table 5-21 was revised to provide the total dose increases for consistency. The corrections to the Class 1, 3a, and 3b calculations were shown on Tables 5-13, 5-21 and 6-1 of Enclosure 1 of the licensee's response to the RAIs in the letter dated November 21, 2016. The dose rate increase from 3-per-10 years to 1-per-10 years is 5.08E-3; the dose increase from 3-per-10 years to 1-per-15 years is 8.71 E-3; the dose rate increase from 1-per-10 years to 1-per-15 years is 3.63E-3. These changes were only minor corrections and have no impact on the conclusion of the LAR. The reported increase in total population dose is below the values provided in EPRI TR-1009325, Revision 2, and defined in Section 3.2.4.6 of the final SE for Revision 2 of NEI 94-01. Thus, this increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

3. The increase in CCFP due to change in test frequency from three in 10 years to once in 15 years is 0.824 percent. This value is below the acceptance guidelines in Section 3.2.4.6 of the final SE for Revision 2 of NEI 94-01.

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

3.5.3 Leak Rate for the Large Preexisting Containment Leak Rate Case

The third condition stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2-A, acceptable, the average leak rate for the preexisting containment large leak rate accident case (i.e., Accident Case 3b) used by the licensees shall be 100 L_a instead of 35 L_a. As noted by the licensee in the table in Section 3.3.1 of Attachment 1 to the LAR, the methodology in EPRI TR-1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the preexisting containment large leakage rate accident case (Accident Case 3b),

and this value has been used in the JAFNPP specific risk assessment. Accordingly, the third condition is met.

3.5.4 Applicability if Containment Overpressure is Credited for ECCS Performance

The fourth condition stipulates that in instances where containment overpressure is relied upon for ECCS performance, an LAR is required to be submitted. In Section 2 and Section 5.2.4 of Attachment 4 to the LAR, the licensee stated that containment overpressure is not relied upon for ECCS performance for JAFNPP. Accordingly, the fourth condition is met.

3.6 Conclusion

Based on the preceding regulatory and technical evaluations, the NRC staff finds that the licensee has adequately implemented its primary containment leakage rate testing program consisting of ILRT and LLRT. The results of the recent ILRT and LLRT combined totals demonstrate acceptable performance and that the structural and leaktight integrity of the primary containment vessel is adequately managed and will continue to be periodically monitored and managed by the ILRTs and LLRTs. The staff finds that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting Revision 3-A of NEI 94-01 and the limitations and conditions identified in the NRC SE incorporated in Revision 2-A of NEI 94-01. Therefore, the NRC staff concludes that the proposed changes to TS 5.5.6, regarding the primary containment leakage rate testing program, are acceptable and will continue to meet the requirements of 10 CFR 50.36.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York 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 October 11, 2016 (81 FR 70178). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be

conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Entergy Nuclear Operations, Inc. letter from Brian R. Sullivan to USNRC, "License Amendment Request to Revise Technical Specifications Section 5.5.6 for Extension of Type A and Type C Leak Rate Test Frequencies," dated August 29, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16242A332).

2. Energy Nuclear Operations, Inc., letter from Brian R. Sullivan to USNRC, "Response to Request for Additional Information (RAI) Regarding License Amendment Request to Revise Technical Specifications Section 5.5.6 for Extension of Type A and Type C Leak Rate Test Frequencies – Supplement 1," dated November 21, 2016 (ADAMS Accession No. ML16326A444).

3. Nuclear Energy Institute Topical Report, NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," August 2012 (ADAMS Package Accession No. ML122210254).

4. Nuclear Energy Institute Topical Report, NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008 (ADAMS Accession No. ML100620847).

5. Staff Requirements Memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736).

6. License Amendment No. 279, "James A. Fitzpatrick Nuclear Power Plant – Amendment Re: One-Time Extension of Containment Integrated Leakage Rate Test Interval," dated September 28, 2004 (ADAMS Accession No. ML042720209).

7. Final Safety Evaluation for NEI 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated June 25, 2008 (ADAMS Accession No. ML081140105).

8. Final Safety Evaluation for NEI 94-01, Revision 3, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated June 8, 2012 (ADAMS Accession No. ML121030286).

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

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

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

12. Generic Issue (GI) 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants, Safety/Risk Assessment," August 2010 (ADAMS Accession No. ML100270582).

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Date: January 24, 2017

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

Sincerely,

/RA/

Diane L. Render, Ph.D.
Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures:
1. Amendment No. 312 to DPR-59
2. Safety Evaluation

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