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

February 3, 2015

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: EXTENSION OF TYPE A PRIMARY CONTAINMENT INTEGRATED LEAK RATE TEST FREQUENCY TO 15 YEARS (TAC NO. MF3279)

Dear Sir or Madam:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 252 to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 20, 2013, as supplemented by letters dated March 11, September 2, October 28, December 3, December 23, 2014, and January 15, 2015.

The amendment extends the 10-year frequency of the ANO-1 Type A primary containment integrated leak rate testing required by TS 5.5.16, "Reactor Building Leakage Rate Testing Program," to 15 years on a permanent basis. The amendment also revises TS 5.5.16 by incorporating Nuclear Energy Institute (NEI) topical report NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," Revision 2-A, as the implementation document for the ANO-1 performance-based leakage rate testing program in accordance with Option B of Appendix J to 10 CFR.

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,

Andrea E. George, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Amendment No. 252 to DPR-51
2. Safety Evaluation

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 252
Renewed License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated December 20, 2013, as supplemented by letters dated March 11, September 2, October 28, December 3, and December 23, 2014, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

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-51 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 252, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance. As part of the implementation of the amendment, the licensee shall incorporate its commitment to use the NEI 94-01, Revision 2-A, Section 5.0 definition for calculating the Type A leakage rate into the ANO-1 Safety Analysis Report (SAR) in the next periodic update in accordance with 10 CFR 50.71(e), as described in the NRC staff's safety evaluation for this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric R. Oesterle, Acting Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-51
and Technical Specifications

Date of Issuance: February 3, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 252

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Renewed Facility Operating License No. DPR-51 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.

Operating License

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
5.0-19	5.0-19

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.16 Reactor Building Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the reactor building 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. The next Type A test performed after the December 16, 2005 Type A test shall be performed no later than December 16, 2020.

In addition, the reactor building purge supply and exhaust isolation valves shall be leakage rate tested once prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

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

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at ≥ 10 psig.

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

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

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

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

(1) Maximum Power Level

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 252, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

(3) Safety Analysis Report

The licensee's SAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 14, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than May 20, 2014.

(4) Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Physical Security Plan, Training and Qualifications Plan, and Safeguards Contingency Plan," as submitted on May 4, 2006.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 252 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By application dated December 20, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13358A195), as supplemented by letters dated March 11, September 2, October 28, December 3, and December 23, 2014 (ADAMS Accession Nos. ML14070A399, ML14245A232, ML14301A590, ML14337A709, and ML14357A437, respectively), Entergy Operations, Inc. (Entergy, the licensee), requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit 1 (ANO-1). The proposed changes would revise TS 5.5.16, "Reactor Building Leakage Rate Testing Program," by replacing the reference to U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (ADAMS Accession No. ML003740058), with a reference to Nuclear Energy Institute (NEI) Topical Report (TR) NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," dated October 2008 (ADAMS Accession No. ML100620847), as the implementation document used by the licensee to develop the ANO-1 performance-based leakage testing program in accordance with Option B of 10 CFR Part 50, Appendix J.

In accordance with the guidance in NEI 94-01, Revision 2-A, the proposed change would permit the performance-based primary containment Integrated Leak Rate Testing (ILRT), also known as a Type A test, interval to be extended from no longer than 10 years to no longer than 15 years, with the next Type A test at ANO-1 to be performed no later than December 16, 2020.

The supplemental letters dated September 2, October 28, December 3, and December 23, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 1, 2014 (79 FR 18331).

Enclosure

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.54(o) require 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 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 Appendix J.

The testing requirements in Appendix J ensure that leakage through the primary containment and related systems and components penetrating primary containment do not exceed allowable leakage rate values specified in the TSs, and that integrity of the containment structure is maintained during its service life.

The licensee has adopted Option B for meeting the requirements of Appendix J. Option B of Appendix J 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. After the preoperational tests, these tests are required to be conducted at periodic intervals based on the historical performance of the overall containment system (for Type A tests), and based on the safety significance and historical performance of each boundary and isolation valve (for Type B and C tests) to ensure integrity of the overall containment system as a barrier to fission product release. The leakage rate test results must not exceed the allowable leakage rate 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.

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 technical specifications.

The implementation document that is currently referenced in the ANO-1 TS 5.5.16, "Reactor Building Leakage Rate Testing Program," is RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. 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 26, 1995, as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B to Appendix J to 10 CFR Part 50, subject to four regulatory positions delineated in Section C of the RG. NEI 94-01, Revision 0, includes provisions that allow the performance-based Type A test interval to be extended to up to 10 years, based upon two consecutive successful tests.

NEI 94-01, Revision 2-A, describes an approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J. 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 safety evaluation (SE) dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC staff concluded that NEI 94-01, Revision 2-A, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TSs in regards to containment leakage rate testing, subject to the specific limitations and conditions listed in Section 4.1 of the SE.

NEI 94-01, Revision 3-A, has also been reviewed by the NRC and approved for use. The final SE for Revision 3, issued by letter dated June 8, 2012 (ADAMS Accession No. ML121030286), includes two specific limitations and conditions listed in Section 4.0 of the SE for Type C test. However, the licensee is not pursuing a Type C extension at this time. This ANO-1 submittal will be reviewed with the conditions and limitations presented in the NRC SE for Revision 2-A of NEI 94-01.

In accordance with the guidance in NEI 94-01, Revision 2-A, the licensee proposes to extend the containment Type A test interval from the current approved 10 years to 15 years, based on acceptable performance. This would allow the next Type A test to be performed within 15 years from the last test (December 16, 2005), instead of the current 10 year interval.

The ANO-1 Reactor Building Leakage Rate Testing Program will continue to comply with the requirements of 10 CFR Part 50, Appendix J..

3.0 TECHNICAL EVALUATION

As described in Section 5.2.1.1 of the ANO-1 Safety Analysis Report (SAR), the reactor building is a reinforced pre-stressed concrete structure in the shape of a cylinder with a shallow dome roof and a flat foundation slab. The circular cylinder wall is pre-stressed by a system of horizontal and vertical tendons. The horizontal tendons are anchored at three buttresses equally spaced around the outside of the containment and the vertical tendons are anchored to the base slab at the bottom and the ring girder at the top. The dome has a 3-way post-tensioning system. The foundation slab is conventionally reinforced with high strength reinforcing steel and is founded on bedrock. A continuous access gallery is provided beneath the base slab for installation of vertical tendons. A welded steel liner is attached to the inside face of the concrete shell to insure leak tightness. The base slab liner plate is installed on top of the structural slab and is covered with concrete.

The ANO-1 leak-tight integrity of the penetrations and isolation valves are verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak-tight integrity and structural integrity of the primary containment is verified through a Type A ILRT, as required by 10 CFR 50, Appendix J. The leakage rate testing requirements of 10 CFR 50 Appendix J Option B (Type A, Type B, and Type C Tests) and the containment in-service inspection (CISI) requirements mandated by 10 CFR 50.55a, together, ensure the continued leak-tight and structural integrity of the containment during its service life.

3.1 Proposed TS Changes

ANO-1 TS 5.5.16, "Reactor Building Leakage Rate Testing Program," currently states, in part:

> A program shall be established to implement the leakage rate testing of the reactor building 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 April 16, 1992 Type A test shall be performed no later than April 15, 2007.

In its application, the licensee requested to revise its TS to implement NEI 94-01, Revision 3-A, into ANO-1 TS 5.5.16 as the guidance document for its performance-based containment leakage rate testing program. Based on clarification requests from the NRC staff, the licensee supplemented its application in a letter dated October 28, 2013, to include the conditions and limitations in Section 4.1 of the NRC staff's SE for NEI 94-01, Revision 2-A, as they were inadvertently omitted from NEI 94-01, Revision 3-A. Based upon further discussion, the NRC staff advised Entergy that the NEI 94-01, Revision 2-A was the appropriate reference for extending the Type A testing frequency and that referencing NEI 94-01, Revision 3-A is appropriate for an extension of Type B and C testing frequencies. The licensee submitted a supplement to its application on December 23, 2014, reflecting the appropriate NEI 94-01 revision, NEI 94-01, Revision 2-A, in its updated proposed TS change, which stated:

> A program shall be established to implement the leakage rate testing of the reactor building 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. The next Type A test performed after the December 16, 2005 Type A test shall be performed no later than December 16, 2020.

To extend the Type A test interval, the 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's evaluation of ANO-1's Type A test performance history, the licensee's ability to meet NEI 94-01, Revision 2-A, Section 9.2.3 requirements, and an evaluation of the licensee's compliance with the six conditions listed in Section 4.1 of the NRC staff's SE for NEI 94-01, Revision 2-A are presented in the following subsections.

3.2 Historical Plant-Specific Containment Leakage Testing Program Results

3.2.1 Last Two Type A Test Results

In the license amendment request (LAR) Section 4.1, the licensee presented the results of the last two Type A tests at ANO-1, which are summarized in Table 1 below.

Table 1: ANO-1 Last Two Type A Test Results

Test Completion Date	Test Pressure, pounds per square inch, gauge (psig)	Performance Leakage Rate (% of containment air weight per day)	
		As-Found (< 0.2 wt%/day for acceptable)	As-Left (< 0.15 wt%/day for acceptable)
April 16, 1992	59.8	0.1308 (4) 0.0904 (5)	0.1245 (1) 0.0841 (2)
December 16, 2005	59	0.0713 (3)	0.0696 (3)

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

As indicated in the LAR and the ANO-1 TS, the maximum allowable containment leakage rate, L_a, is 0.2 percent of containment air weight per day at the peak calculated containment internal pressure for design basis loss-of-coolant accident, P_a.

In Section 4.1 of the LAR, the licensee stated that a total of six Type A tests have been performed at ANO-1. The licensee also provided results of the April 16, 1992, Type A test, where the as-left leakage rate was calculated at 0.1245 weight percent per day (wt%/day), which was less than the allowable leak rate of 0.150 wt%/day; and the results of the December 16, 2005, Type A test, which was conducted at 58 pounds per square inch gauge (psig) test pressure (design pressure is 59 psig) with the as-found and the as-left containment leakage rates calculated at 0.0713 wt%/day and 0.0696 wt%/day, respectively, which were less than the acceptance criterion of 0.75 L_a, or 0.15 wt%/day.

The NRC staff notes that the results for the 1992 test in Table 1 are based on 10 CFR 50 Appendix J, Option A, which was the approved methodology at the time, while Option B was adopted by the licensee in 1996. The 1992 test results were reviewed by the NRC staff in accordance with the methodology in use by the licensee at the time, to establish the performance history as described in Section 9.2.3 of NEI 94-01, Revision 2-A. In addition, the as-found leakage rates as indicated in Notes (4) and (5) are based on Total Time and Mass Point methods, respectively. These leakage rates excluded the leakage from the Reactor Building Purge Penetration V-1 that was found with a large leakage rate. In its supplement dated December 3, 2014, in response to an NRC staff request for additional information (RAI) dated November 19, 2014 (ADAMS Accession No. ML14323A745), the licensee explained that the Reactor Building Purge Penetration V-1, after the local leakage rate adjustments specified by NRC Information Notice 85-71, "Containment Integrated Leak Rate Tests," dated August 22, 1985 (ADAMS Accession No. ML031180640), was not able to maintain test pressure between Valves CV-7402 and CV-7404 during the Type C test. After repairs (replacement of 35 seal adjustment bolts on Valves CV-7402 and CV-7404) the as-left leakage was not detectable. Based on Section 9.2.3 of NEI 94-01, Revision 2-A, since the V-1 penetration leakage was able to be determined by a local leakage rate test and found an undetectable as-left leakage rate, the resultant undetectable leakage would leave the performance leakage rate reported in Notes (4) and (5) unchanged.

The results in Table 1 indicate that the two most recent consecutive Type A tests at ANO-1 conducted at test pressure (calculated peak accident pressure (P_a)) were successful with containment performance leakage rate less than the maximum allowable containment leakage rate (L_a at P_a) of 0.2 percent containment air weight per day. The NRC staff concludes, based on the information provided by the licensee regarding containment leakage rate historical data, and because the licensee will be required by TSs to use the definition for calculating the Type A test performance leakage rate to demonstrate leakage integrity and to determine extended Type A test intervals in Section 5.0 of NEI 94-01, Revision 2-A (see NRC Condition 1 discussion in Section 3.5.1 of this SE), that the successful completion of the two most recent periodic Type A tests supports extending the current ANO-1 ILRT interval to 15 years.

3.2.2 Type B and Type C Testing Program

The licensee described its Type B and Type C Testing Program in Section 4.2 of its 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 Section 4.2 of the LAR, the combined Type B and Type C leakage rate test results for both as-found condition (on minimum pathway basis) as well as for as-left condition (on maximum pathway basis) for the most recent tests performed in 1R22, 1R23 and 1R24 for ANO-1 were provided. The results indicate that the combined as-found minimum path leak rate from the Type B and Type C tests did not exceed the acceptance criterion.

In the Attachment 5 to the LAR, the licensee provided a summary of individual components that did not meet the administrative limit for Type B and Type C testing. In a September 2, 2014, response to an NRC staff RAI dated July 31, 2014 (ADAMS Accession No. ML14209A085), the licensee provided the causes and corrective actions taken to address these components that failed the administrative limit. The NRC staff reviewed the licensee's response and concludes that the failure causes were identified and appropriate corrective actions were taken.

The licensee also provided the maximum and minimum pathway leakage rate for the last three refueling outages as shown in Table 2 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 199,663 standard cubic centimeters per minute (sccm).

Table 2: Pathway Leakage Values for the Last Three ANO-1 Refueling Outages

Refueling Outage	Leakage	Value, in sccm
1R22 (2010)	As-Found Minimum Pathway Leakage	11,642
	As-Left Maximum Pathway Leakage	26,193
1R23 (2011)	As-Found Minimum Pathway Leakage	19,734
	As-Left Maximum Pathway Leakage	35,601
1R24 (2013)	As-Found Minimum Pathway Leakage	10,361
	As-Left Maximum Pathway Leakage	19,599

The licensee stated that industry experience has shown that the Type B and C tests can identify the vast majority (over 95 percent) of all potential primary containment leakage paths. The licensee stated that its LAR adopts the guidance in NEI 94-01, Revision 3-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, and that Type B and Type C testing will continue to provide a high degree of assurance that primary containment integrity is maintained. In its supplement dated December 23, 2014, the license revised its application (and associated TS changes) to adopt the guidance of NEI 94-01, Revision 2-A, since the licensee is only requesting to extend the Type A test frequency with this LAR.

In an RAI dated July 31, 2014, the NRC staff requested information regarding the penetrations and components at ANO-1 subject to Type B and Type C testing. In its September 2, 2014, response to the RAI, the licensee stated the following:

a) ANO-1 has 46 penetrations with 55 components that are Type B tested on a 120-month frequency with the exception of access hatches (four penetrations/ nine components) that are tested on a 30-month frequency. The other exception is that three of the Type B tested components are part of Type C penetrations and are tested on a 60-month frequency. Currently no Type B-tested component that is on the extended test frequency (60- or 120-month) is on the accelerated 30-month test frequency.

b) ANO-1 has 37 penetrations with 104 components (including the three Type B tested components discussed above) that are Type C tested on a 60-month frequency with the exception of purge valves (two penetrations/four components) that are tested on a 30-month frequency.

c) If one of the components that is on the 120-month (or 60-month) frequency fails an LLRT then it is placed on a 30-month frequency until it has passed two consecutive as-found LLRTs at which time it is returned to 120-month (or 60-month) test frequency.

In Attachment 5 to the LAR, the licensee provided a summary table of components that did not meet the administrative limit, as found in the ANO-1 operations or engineering procedure, for Type B and Type C testing since 2002. In an RAI dated July 31, 2014, the NRC staff requested information regarding causes and corrective actions taken for those components listed in Attachment 5 to the LAR. In its September 2, 2014, response to the RAI, the licensee described the causes and corrective actions taken to address the components that did not demonstrate acceptable performance by exceeding the LLRT administrative leak rate limit.

Based on information provided by the licensee and the evaluation above, the NRC staff concludes that (1) the performance history of successful completion of two most recent consecutive periodic Type A tests supports extending the current ILRT interval to 15 years; (2) the combined leakage from the Type B and Type C tests has been consistently maintained well below the acceptance criteria; (3) a majority of penetrations that are subject to performance based Type B or Type C test are 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 ANO-1; and (4) the licensee has appropriately taken corrective actions and has adjusted test schedules consistent with its Appendix J, Option B program, for those components that failed their allowable administrative limits.

Based on the information discussed above, the NRC staff concludes that the licensee has provided information to assure that the it is appropriately implementing its Type B and Type C Testing program under Option B, in a rational and systematic manner that is consistent with the implementation document in the TS, and will continue to do so, in accordance with NEI 94-01, Revision 2-A, if the current ILRT interval is extended to 15 years. Thus, 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 proposed for reference in the ANO-1 TSs.

3.3 Containment In-Service Inspection Program (American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Subsections IWE and IWL)

In Section 4.0 of the LAR, the licensee stated that general visual examinations of accessible interior and exterior surfaces of the containment system for structural problems are conducted in accordance with the ANO-1 Containment In-service Inspection (CISI) plan and schedule, which implements the requirements of the ASME Code, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g). The IWE/IWL inspections and supplemental inspections, in accordance with other approved plant procedures, are used to satisfy the general visual examination requirements of Appendix J, Option B and to monitor and manage the age-related degradations of the primary containment to ensure that containment structural and leak-tight integrity is maintained through its service life.

In Attachment 4 to the LAR, the licensee provided a summary of ANO-1 containment building inspection results, including Table 4-1 for the results of the post-tensioning tendon surveillance and IWL containment building inspection, Table 4-2 for the results of the ANO-1 containment building interior and exterior structural inspection surveillances performed during each refueling outage and prior to any integrated leak test, and Table 4-3 for the IWE containment building inspection results. In Attachment 4 to the LAR, the licensee stated that the ANO-1 CISI plan has three periods during each 10 year interval, and that ANO-1 performs a dome inspection in the first outage in a period and a barrel inspection during the next outage in the period. This periodicity is repeated for the next period. If a period only has one outage, then ANO-1 will perform both a barrel and dome inspection.

The equipment hatch augmented examination and the examination of the moisture barrier at the interface of the concrete base floor and the containment wall liner plate is discussed in Section 3.5.3 of this SE and will not be repeated here.

In its LAR, as supplemented by letters dated September 2 and October 28, 2014, the licensee provided information regarding the results of IWE/IWL inspections performed since the last Type A test performed in 2005, and corrective actions taken to disposition them. The licensee stated that no flaws or areas of degradation exceeding the allowable acceptance standards of IWE-3500 or IWL-3200 were identified.

In Table 4-3 of Attachment 4 of the LAR, the licensee stated that (1) visual and ultrasonic inspections were performed on the containment wall liner plate in May 2007, April 2010, and May 2013; (2) visual inspections were performed on the containment dome liner plate in October 2005, November 2008, and October 2011; (3) the containment liner plate inspection findings included minor rust with no visible pitting and blistering and flaking of the liner plate topcoat; (4) the prime coat of the liner plate coating system was found intact providing full protection to the liner plate; and (5) none of the deficiencies found during these inspections affected the liner plate structural integrity and the conditions were acceptable without engineering evaluation or repair.

In an RAI dated July 31, 2014, the NRC staff requested information regarding any instances where existence of or potential for degraded conditions in inaccessible areas of the concrete containment structure and steel liner were identified and evaluated based on 10 CFR 50.55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A). The licensee stated, in its RAI response dated September 2, 2014, that there were three indications in the 35th year (May 2009) and four indications in the 40th year (May 2013) IWL inspections of the exterior of the containment and the containment post-tensioning system that required evaluation under ASME Code, IWL-3300 and were found acceptable. The licensee also indicated that no indications were found that challenged structural integrity or leak tightness of the containment.

Based on the information provided by the licensee detailing the results of the recent IWE/IWL inspections discussed above, the NRC staff concludes that there has not been evidence to date of significant degradation of the ANO-1 containment structure, and the degradations noted have been entered into the ANO-1 corrective action program and appropriately managed and/or corrected. Based on the above evaluation, the NRC staff concludes that the licensee is adequately implementing its CISI program to monitor and manage age-related degradation of the ANO-1 containment structure, and that the program is adequate to support an extension of the ILRT frequency to 15 years at ANO-1.

3.4 Probabilistic Risk Assessment

Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of NEI 94-01, Revision 2-A, states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," of NEI 94-01 states that the assessment should be performed using the approach and methodology described in Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2-A[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 its SE dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC staff found the methodology in EPRI TR-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 NRC SE for EPRI TR-1009325, Revision 2, stipulate that:

1. The licensee submits documentation indicating that the technical adequacy of its Probabilistic Risk Assessment (PRA) is consistent with the requirements of Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," 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 NRC SE for EPRI TR-1009325, Revision 2.

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

3.4.1 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval from 10 years to 15 years. The risk analysis was provided in Attachment 6 of the licensee's LAR. Additional information was provided by the licensee in its letter dated September 2, 2014, in response to NRC RAIs.

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

[2] The safety evaluation report for EPRI TR-1009325, Revision 2, indicates that the clarification regarding small increases in risk is provided in Section 3.2.4.5; however, the clarification is actually provided in Section 3.2.4.6.

In Section 4.5.1 of Attachment 1 to the LAR, the licensee stated that the plant-specific risk assessment follows the guidance in NEI 94-01, Revision 2-A, the methodology described in EPRI TR-1009325, Revision 2-A, and the NRC regulatory guidance outlined in RG 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated November 2002 (ADAMS Accession No. ML023240437). In Section 5.1 of Attachment 6 to the LAR, the licensee stated that in order to incorporate the containment liner corrosion issue into its ILRT extension risk evaluation, it utilized the Calvert Cliffs Nuclear Plant (CCNP) methodology, as modified by EPRI. In an RAI dated July 31, 2014 (RAI 5), the NRC staff requested more information regarding more recent liner corrosion instances since the CCNP methodology was performed in 2002. In the licensee's response to RAI 5, included in the supplement dated September 2, 2014, the licensee provided corrosion sensitivity results obtained from the CCNP methodology using a data collection period spanning from September of 1996 to December 31, 2013. Furthermore, the licensee performed an assessment of the impact of external events by calculating increase in large early release frequency (LERF) (EPRI Accident Class 3b frequency) due to seismic and internal fire.

The licensee addressed each of the four conditions for the use of EPRI TR-1009325, Revision 2, which are listed in Section 4.2 of the NRC SE for that report. A summary and evaluation of how the licensee has met each condition is provided in the following subsections.

3.4.1.1 Technical Adequacy of the PRA

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

In Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (ADAMS Accession No. ML070650428), the NRC staff clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200 (ADAMS Accession No. ML070240001) to assess the technical adequacy of the PRA used to support risk-informed applications. Revision 2 of RG 1.200 (ADAMS Accession No. ML090410014) will be used for all risk-informed applications received after March 2010. In Section 3.2.4.1 of the NRC staff SE for EPRI TR-1009325, Revision 2, the NRC staff states that Capability Category I of the ASME PRA standard shall be applied as the standard for assessing PRA quality for IRLT extension applications, as approximate values of core damage frequency (CDF) and LERF, and their distribution among release categories, are sufficient to support the evaluation of changes to ILRT frequencies.

In Section 4.5.2 of Attachment 1 to the LAR, the licensee stated that the ANO-1 PRA model is composed of a Level 1 and LERF models for internal events and that severe accident sequences have been developed from internally initiated events. The sequences have been mapped to the radiological release end state (i.e., source term release to environment). The licensee also stated that the ANO-1 PRA was initially developed in response to the NRC Generic Letter (GL) 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities – 10 CFR § 50.54(f)," dated November 23, 1988 (ADAMS Accession No. ML031470299), for Individual Plant Examinations (IPE) which underwent NRC review. The licensee stated that the review comments, current plant design, current procedures, plant operating data, current industry PRA techniques, and general improvements identified by the NRC have been incorporated into the current PRA model. The licensee further stated that the model is maintained in accordance with Entergy PRA procedures.

In Section 4.5.2 of Attachment 1 to the LAR, the licensee stated the ANO-1 PRA internal events model has been upgraded to meet the guidance in RG 1.200, Revision 1. In an RAI (RAIs 1 and 2) dated July 31, 2014, the NRC staff requested information pertaining to the ANO-1 PRA internal events model, including the scope of the PRA peer review, the RG 1.200 revision used for the peer review, and details of the gap analysis. The licensee stated in its response to RAI 1, provided in the supplement dated September 2, 2014, that an industry peer review of the ANO-1 PRA was completed against RG 1.200, Revision 2, in August 2009. Furthermore, in response to RAI 2, in the same supplement, the licensee provided a list of Facts and Observations (F&Os) from the peer review relevant to the submittal and descriptions of how they were addressed or the impact of open items on the application. The list of F&Os includes description of items that are considered closed by the licensee and a number of open items that do not meet ASME standard requirements. The NRC staff reviewed the F&Os related to each PRA attribute and the licensee's disposition or description of impact and concludes that, for the level of PRA quality needed for this licensing action, and considering the large margins between the reported results in the LAR and the RG 1.174 acceptable guidelines (discussed in the next section), the F&Os have either been adequately addressed, do not have a significant impact on risk evaluations for this application, or are only documentation issues.

In Section 5.3 of Attachment 6 to the LAR, the licensee provided information regarding an analysis of the external events contributions to the PRA and the results complied with the guidelines for risk increase in RG 1.174. The licensee also reported the seismic CDF in the LAR, which is within the range of seismic CDF estimates using 2008 U.S. Geological Survey seismic hazard curves. In an RAI (RAI 4) dated July 31, 2014, the NRC staff requested information regarding the Fire PRA submitted in support of the ANO-1 LAR dated January 29, 2014 (ADAMS Accession No. ML14029A438), to adopt the National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection Light Water Reactor Electric Generating Plants," as it pertains to potential impacts on the external events risk evaluation performed in support of the ILRT extension request. In its response to RAI 4, provided in the supplement dated September 2, 2014, the licensee clarified that, as the ANO-1 Fire PRA model was not available at the time of the ILRT analysis, a surrogate was developed to estimate the potential risk contribution due to fire-induced initiating events. Also in its response to RAI 4, the licensee provided an updated risk analysis based on the current CDF for the ANO-1 FPRA, and stated that while the inclusion of the FPRA results did cause an increase in the combined internal and external events LERF, the criteria of RG 1.174 remains satisfied.

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

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT

intervals." This section also states that: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [(i.e., EPRI Report No. 1009325, Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed." This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval."

Therefore, based on information provided by the licensee, the NRC staff concludes that the information used by the licensee to estimate the effect on total LERF due to external events is acceptable.

Given that the licensee has evaluated its PRA against ASME/ANS RA-Sa-2009, along with Revision 2 of RG 1.200, evaluated the findings developed during the reviews of its PRA for applicability to the ILRT extension, and either addressed the findings or explained their impact, the NRC staff concludes that ANO-1 PRA model used for this application is of sufficient technical adequacy to support the risk evaluation of extending the ANO-1 ILRT interval to 15 years. Accordingly, based on information provided by the licensee and the evaluations above, the NRC staff concludes that the first condition is met.

3.4.1.2 <u>Estimated Risk Increase</u>

The second condition in Section 4.2 of the NRC SE for EPRI TR-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-A. Specifically, the clarification in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2-A, indicated that a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-roentgen equivalent man (rem) per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points. Additionally, for plants that rely on containment over-pressure for net positive suction (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. Thus, the associated risk metrics include: CDF, LERF, population dose, and CCFP.

The licensee reported the results of the plant-specific risk assessment in Section 4.5.3 of Attachment 1 to the LAR and in responses to NRC staff RAIs dated July 31, 2014 (RAIs 3, 4, and 5) provided in the supplement dated September 2, 2014. Details of the risk assessment are provided in Attachment 6 to the LAR. The licensee reported risk impacts for a change in the Type A ILRT test interval from a three-per-10-years (the test frequency under 10 CFR 50 Appendix J, Option A) to a once per-15-years (risk impact from baseline) and from a once per-10-years to a once per-15-years (risk impact from current). The following is a summary of the licensee's analysis associated with extending the Type A ILRT frequency:

- The reported increases in LERF for internal events from baseline and current are 2.63E-8 per year and 1.09E-8 per year, respectively. The increase in LERF from baseline for internal and external events combined, after including the ANO-1 FPRA results in response to RAI 4 provided in the supplement dated September 2, 2014, is reported as 2.29E-7 per year. This increase in LERF is considered small according to the RG 1.174 acceptance guidelines. As the total LERF for ANO-1 reported in the licensee's response to RAI 4 and the LAR to adopt NFPA-805 is less than 1E-5 per year, the small increase in LERF reported in response to RAI 4 is acceptable according to RG 1.174 acceptance guidelines.

- The change in CCFP is reported to be 3.77E-3 (0.38%) for the current 10-year interval to a 15-year interval and 9.04E-3 (0.90%) for the cumulative change of going from a three-per-10-year interval to a once per-15-year interval. This is less than the acceptance guideline value of 1.5 percentage points for a small increase in CCFP, as provided in EPRI TR-1009325, Revision 2-A, and defined in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2.

- The increases in population dose from extending the current ten-year interval and the original three-per-10-year to a 15-year interval are reported as 1.70E-4 person-rem per year and 4.08E-4 person-rem per year, respectively. The increase in population dose rate from extending the current 10-year interval to a 15-year interval was reported as 2.3E-4 person-rem per year, when the licensee used the Severe Accident Mitigation Alternative (SAMA) population data in response to RAI 3, provided in the supplement dated September 2, 2014. The increase in population dose is less than the values associated with a small increase, as provided in EPRI TR-1009325, Revision 2-A.

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, the increase in the total integrated plant risk and the magnitude of the change in the CCFP for the proposed change are small and supportive of the proposed change. 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, based on information provided by the licensee and the evaluation above, the NRC staff concludes that the second condition is met.

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

The third condition in Section 4.2 of the NRC SE for EPRI TR-1009325, Revision 2, stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2, acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a.

As noted by the licensee in Section 4.5.1 of Attachment 1 to the LAR, the methodology in EPRI TR-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 ANO-1 plant-specific risk assessment. Based on the information provided by the licensee, the NRC staff concludes that the third condition is met.

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

The fourth condition in Section 4.2 of the NRC SE for EPRI TR-1009325, Revision 2, stipulates that in instances where containment over-pressure is relied upon for ECCS performance, an LAR is required to be submitted. In Section 4.5.1 of Attachment 1 to the LAR, the licensee stated that ANO-1 does not rely on containment over-pressure to assure adequate net positive suction head for ECCS pump following design-basis accidents. Based on information provided by the licensee, the NRC staff concludes that the fourth condition is met.

3.5 Adoption of NEI 94-01, Revision 2-A

The NRC staff's primary review method was to ensure the six limitations and conditions as set forth in the NRC staff's SE dated June 25, 2008, endorsing NEI 94-01, Revision 2, were met. In its SE, the NRC staff concluded that NEI 94-01, Revision 2-A, describes an acceptable approach for implementing the optional performance-based requirements of Option B of Appendix J to 10 CFR Part 50, and is acceptable for referencing by licensees proposing to amend their TSs regarding containment leakage rate testing, subject to the specific limitations and conditions listed in Section 4.1. Section 3.1 of the staff's SE provides the NRC staff position on the adequacy of NEI 94-01 in addressing the performance-based Type A, Type B, and Type C test frequencies. It also addresses the adequacy of pre-test inspections, procedures to be used after major modifications to the containment structure, deferral of tests beyond the 15-year interval, and the relation of CISI requirements mandated by 10 CFR 50.55a to the containment leak rate testing requirement.

The NRC staff evaluated whether the licensee adequately addressed and satisfied the six limitations and conditions in staff SE dated June 25, 2008, as discussed below.

3.5.1 NRC Condition 1

Condition 1 of the NRC staff's SE for NEI 94-01, Revision 2-A states: "For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2-A, in lieu of that in [American National Standards Institute/American Nuclear Society (ANSI/ANS)]-56.82002. (Refer to SE Section 3.1.1.1)."

In a table presented in Section 4.0 of the LAR (Reference 1), 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 ANO-1 Type A tests are performed. Because the licensee's proposed TS change incorporates NEI 94-01, Rev. 2-A as the guidance document for the containment leakage rate program, the licensee will be required to follow the definition described above in order to maintain compliance with the TSs. Therefore, the NRC staff concludes that the licensee has adequately addressed Condition 1.

3.5.2 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 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 has been extended to 15 years."

In response to clarification requests from the NRC staff regarding the licensee's responses to RAIs 7 and 8 (in the licensee's supplement dated September 2, 2014), the licensee provided updated RAI 7 and 8 responses in a supplement dated October 28, 2014. In this supplement, the licensee stated that the examinations performed in accordance with the ANO-1 ASME Code, Section XI, Subsection IWE/IWL program satisfy the general visual examination requirements specified in 10 CFR 50, Appendix J, Option B. Also, in its supplement dated October 28, 2014, the licensee stated that the frequency of examinations per Subsection IWE (three examinations over a 10-year interval) assures that at least three general visual examinations of metallic components will be conducted between the Type A tests and one scheduled immediately before the next Type A test, if the Type A test interval is extended to 15 years.

In its supplement dated October 28, 2014, the licensee also stated that visual examinations of accessible concrete surfaces performed in accordance with Subsection IWL at a frequency of 5 years will result in three IWL examinations being performed during a 15-year ANO-1 Type A test interval; and, in addition to the IWE and IWL examinations, ANO-1 performs a visual inspection of the accessible interior and exterior concrete surfaces of the ANO-1 Containment structure prior to any Type A test. This examination is performed in accordance with approved plant procedures to satisfy the requirements of the 10 CFR 50, Appendix J testing program, and in sufficient detail to identify any evidence of deterioration which may affect the structural integrity or leak tightness of the containment building. The areas that are inspected include the external surface of the containment building, the tendon access area, the basement of the containment building, the wall inside the main steam safety enclosure, and the interior liner plate surface of the containment building. The examinations of the inside of the containment building are performed during Mode 5 or 6. The exterior portion of the inspection may be performed during any mode but must be performed following any repair or modification to the containment building and prior to pressurization of the building. The licensee concluded that, together, these examinations assure that at least four general visual examinations of concrete containment surfaces will be conducted before the next Type A test if the Type A test interval is extended to 15 years.

Additionally, in its supplement dated October 28, 2014, the licensee provided an approximate schedule, shown in Table 3 below, for general visual examinations of containment surfaces, representative of a typical 15-year period between Type A tests.

Table 3: Schedule of IWE/IWL General Visual Examinations

Calendar Year (Outage)	Type A Test (ILRT)	General Visual Examination of Accessible Exterior Surface	General Visual Examination of Accessible Interior (Liner) Surface
2005 (1R19)	X	X	X
2006			
2007 (1R20)			X
2008 (1R21)		X	
2009			X
2010 (1R22)			
2011 (1R23)			X
2012			
2013(1R24)		X	X
2014			
2015(1R25)			X
2016(1R26)			X
2017			
2018 (1R27)		X	X
2019 (1R28)	X	X	X

On the basis that the licensee's schedule of general visual examinations described above results in at least three examinations between Type A tests and one examination immediately prior to the Type A test for both containment concrete and metallic liner surfaces, the NRC staff concludes that the licensee's inspection schedule plan, noted in the LAR and the supplemental information, meets the general visual examination requirements in Section 9.2.3.2 of NEI 94-01, Revision 2-A, and 10 CFR Part 50, Appendix J, Option B, and therefore, satisfies Condition 2 in the NRC staff SE for NEI 94-01, Revision 2-A.

3.5.3 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 Section 4.0 of the LAR, and in its supplement dated September 2, 2014, the licensee stated that general visual examinations of accessible interior and exterior surfaces of the containment system for structural problems are conducted in accordance with the ANO-1 CISI program and schedule, which implements the requirements of the ASME Code, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g); and, the ANO-1 containment system employs moisture barriers but does not employ bellows on penetrations through the containment pressure retaining boundaries.

In Table 4-3 of Attachment 4 to its LAR, the licensee stated that the moisture barrier at the interface of the concrete base floor and the containment wall liner plate was inspected in October 2005, November 2008, and October 2011. In an RAI dated July 31, 2014, the NRC staff requested information regarding any findings or deficiencies from ANO-1 CISI program, including examinations of any moisture barriers. The licensee noted in its supplement dated September 2, 2014, that the moisture barrier examination, performed during the 2011 (1R23) refueling outage, was acceptable with no defects noted.

In response to the NRC staff RAI described above, the licensee, in its letter dated September 2, 2014, stated that during implementation of the ANO-1 CISI program in accordance with ASME Code, Section XI, Subsections IWE/IWL, no areas of potential degraded conditions in inaccessible areas of the concrete containment structure and steel liner have been identified based on conditions found in accessible areas that required evaluation in accordance with 10 CFR 50.55a(b)(2)(ix)(A) and 10 CFR 50.55a(b)(2)(viii)(E).

The licensee, in Section 4.0 of its LAR, stated that there is one primary containment surface associated with the area around the equipment hatch that requires augmented examinations in accordance with the ASME Code, Section XI, IWE-1240.

The NRC staff, in an RAI (RAI 6) dated July 31, 2014, requested information regarding the findings which led to the augmented examinations for the equipment hatch, as well as information on the licensee's methods for monitoring and management of this condition. In response to RAI 6, in its letter dated September 2, 2014, the licensee provided additional information regarding the augmented examination of the equipment hatch. The following is a summary of the information provided by the licensee:

- During an ASME Code, Section XI, IWE general visual examination of the ANO-1 equipment hatch during the 2004 refueling outage, an area of pitting and corrosion on the equipment hatch door and flange area was identified, cleaned, and repainted.

- A review of the examination findings concluded that although the area was acceptable by examination in accordance with IWE-3122.1, the area was subject to accelerated degradation and the requirements of IWE-1240 were to be implemented.

- The ANO-1 CISI program was revised to include augmented examinations of the equipment hatch in accordance with IWE-1240 and Table IWE-2500.

- It was determined that the cause of the pitting and corrosion was repeated wetting and submergence due to trapping storm water against the exterior of the equipment hatch. At the conclusion of each refueling or forced outage, actions are taken to adequately seal the Kelly closure above the equipment hatch to prevent/minimize the intrusion of rain water that can accumulate against the equipment hatch.

- Subsequent to the IWE examination conducted in 2004, augmented examinations of the equipment hatch have been performed in accordance with ASME Code, Table IWE-2500 during the 2005 (1R19), 2010 (1R22), and 2013 (1R24) ANO-1 refueling outages.

- Results of the 1R19 visual examination identified pitting and corrosion on the surfaces of the equipment hatch door and flange area, and subsequently, ultrasonic thickness measurements were taken. An average of 1.0-inch thickness was recorded. As with the 2004 examination, the acceptance criteria of IWE-3122.1 were satisfied and the licensee's operability determination concluded that the structural integrity of the ANO-1 containment building and equipment hatch were not impacted and that the system remained operable. Corrective actions to clean and re-coat the affected surfaces were completed in 2006.

- The satisfactory completion of the augmented examinations conducted during 1R22 and 1R24 refueling outages continued to provide assurance of structural integrity and compliance with the acceptance standards.

- An upcoming augmented examination of the ANO-1 equipment hatch will be conducted during the next refueling outage currently scheduled for 2016 (1R26).

Based on the above information provided by the licensee regarding the augmented examination of the equipment hatch and the ANO-1 operating experience, to date, identifying no conditions that would indicate the presence of any potential degraded conditions in inaccessible areas of the concrete containment structure and steel liner, the NRC staff concludes that the licensee has adequately addressed Condition 3 of the NRC staff SE for NEI 94-01, Revision 2-A.

3.5.4 <u>NRC Condition 4</u>

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)." The NRC staff's intent of requiring licensees to meet Condition 4 when requesting ILRT extensions, with reference to Section 9.2.4 of NEI 94-01, Revision 2-A, is to ensure that licensees clearly understand that following major containment modifications, such as those described in Section 3.1.4 of the NRC SE, the post-repair pressure testing performed must demonstrate both structural and leak-tight integrity of the repaired containment.

In a table presented in Section 4.0 of the LAR, the licensee stated that it had replaced the ANO-1 steam generators and the reactor vessel closure head during December 2005 for which the containment structure required modifications. The licensee stated that a Type A ILRT was completed following containment restoration. The results of the December 2005 ILRT are provided in Section 4.1 of the LAR, with the results indicating that the repairs to the containment adequately met the TS leakage requirements.

In Section 4.1 of its LAR, the licensee also stated that: (1) no modifications that require a Type A test are planned prior to refueling outage 1R28, when the next Type A test will be performed under this proposed change; (2) any unplanned modifications to the containment building 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 reactor building which could have adversely affected reactor building integrity; and (4) there is no anticipated addition or removal of plant hardware within the reactor building which could affect leak-tightness.

Based on the information provided by the licensee, and the post-repair pressure testing (i.e., Type A test) performed by the licensee in 2005 following the major containment modifications for steam generator replacement and the reactor vessel closure head, the NRC staff concludes that the licensee has complied with the NRC staff position with regard to post-repair pressure testing following major containment modifications, as stated in Section 3.1.4 of the NRC SE for NEI 94-10, Revision 2-A, and has no major containment modifications planned before the next proposed ILRT (15-year frequency). Therefore, the NRC staff concludes that the licensee has adequately addressed Condition 4 of the NRC staff SE for NEI 94-01, Revision 2-A.

3.5.5 <u>NRC Condition 5</u>

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 Section 4 of its LAR, that it acknowledges and accepts the NRC staff position in Condition 5, as communicated to the nuclear industry 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," dated December 8, 2008 (ADAMS Accession No. ML080020394). By this, the licensee has confirmed its understanding of the NRC staff position in RIS 2008-27, 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 NRC staff will implement the position in RIS 2008-27 in reviewing such license amendment requests. 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 as clarified in RIS 2008-27. On this basis, the NRC staff concludes that the licensee has adequately addressed Condition 5 of the NRC staff SE for NEI 94-01, Revision 2-A.

3.5.6 <u>NRC Condition 6</u>

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 Report No. 1009325, Revision 2, including the use of past ILRT data."

The licensee stated in the table presented in Section 4.0 of its LAR that this condition is not applicable since ANO-1 is not licensed to 10 CFR Part 52. The NRC staff finds that ANO-1 is currently an operating reactor licensed to 10 CFR Part 50, and therefore, Condition 6 does not apply.

3.5.7 Conclusion – Adoption of NEI 94-01, Revision 2-A

Based on the above evaluation of each condition, the NRC staff concludes 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-A. Therefore, the NRC staff concludes that it is acceptable for ANO-1 to adopt NEI 94-01, Revision 2-A, as the implementation document in its TS 5.5.16, "Reactor Building Leakage Rate Testing Program."

3.6 Conclusion Summary

The NRC staff concludes that the licensee has effectively implemented an adequate containment leakage rate testing (ILRT and LLRT) program, CISI program, and supplemental inspections to periodically examine, monitor, and manage age-related degradation of the ANO-1 primary containment. The results of the past ILRTs, LLRTs, and the CISI program examinations demonstrate acceptable performance of the ANO-1 primary containment and demonstrate that the structural and leak-tight integrity of the primary containment structure is adequately managed. The structural and leak-tight integrity of the ANO-1 primary containment will continue to be periodically monitored and managed by the LLRT and CISI programs, if the current ILRT interval is extended from 10 years to 15 years. Therefore, the NRC staff concludes that there is reasonable assurance that the containment structural and leak-tight integrity will continue to be maintained, without undue risk to public health and safety, if the current ILRT interval at ANO-1 is extended to 15 years.

The NRC staff concludes that the licensee has submitted a PRA of adequate quality in order to evaluate the effect of an ILRT frequency extension to 15 years. Furthermore, the NRC staff concludes, regarding the licensee's risk assessment results, which take into account the proposed ILRT frequency extension, that the increase in LERF is small and consistent with the acceptance guidelines of RG 1.174.

Based on the above evaluations, the NRC staff concludes that the proposed permanent 15-year extension of the Type A containment ILRT for ANO-1 is acceptable. In accordance with the revised TS 5.5.16, the next Type A containment ILRT for ANO-1 shall be performed no later than December 16, 2020.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments. If comments were provided, they should be addressed here.

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. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on April 1, 2014 (79 FR 18331). 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.

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Date: February 3, 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/

Andrea E. George, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Amendment No. 252 to DPR-51
2. Safety Evaluation

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