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1CAN121302

December 20, 2013

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: License Amendment Request
Technical Specification Change to Extend the
Type A Test Frequency to 15 Years
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the following amendment to Arkansas Nuclear One, Unit 1 (ANO-1). The proposed change would allow for the extension to the ten-year frequency of the ANO-1 Type A or Integrated Leak Rate Test (ILRT) that is required by Technical Specification (TS) 5.5.16 to be extended to 15 years on a permanent basis.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that the changes involve no significant hazards consideration. The bases for these determinations are included in the attached submittal.

Similar TS changes were approved for ANO-2 on April 7, 2011 (ML110800034), Palisades on April 23, 2012 (ML120740081), and Nine Mile Point Unit 2 on March 30, 2010 (ML100730032).

The proposed change includes three new commitments. These commitments are summarized in Attachment 7.

Entergy requires approval of the proposed amendment by January 5, 2015. Once approved, the amendment shall be implemented within 30 days.

If you have any questions or require additional information, please contact Stephenie Pyle at 479-858-4704.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 20, 2013.

Sincerely,

ORIGINAL SIGNED BY JEREMY G. BROWNING

JGB/rwc

Attachments:

1. Analysis of Proposed Technical Specification Change
2. Markups of Technical Specification Pages
3. Clean (Revised) Technical Specification Pages
4. Summary of the Results From Reactor Building Inspections
5. List of Components that Failed Type B or Type C Tests Since 2002
6. Risk Analysis
7. List of Regulatory Commitments

cc: Mr. Marc L. Dapas
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Attachment 1 to

1CAN121302

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1).

The proposed amendment revises ANO-1 Technical Specification (TS) 5.5.16, “Reactor Building Leakage Rate Testing Program,” by replacing the reference to Regulatory Guide (RG) 1.163 “Performance-Based Containment Leak-Test Program,” with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, “Industry Guideline for Implementing Performance−Based Option Of 10 CFR Part 50, Appendix J,” as the implementation document used by Entergy Operations, Inc. (Entergy) to develop the ANO-1 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J.

NEI 94-01, Revision 2-A describes an approach for implementing the optional performance-based requirements of Option B, including provisions for extending primary containment integrated leak rate test (ILRT) intervals to 15 years, and incorporates the regulatory positions stated in RG 1.163. In the safety evaluation (SE) issued by NRC letter dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE.

By letter dated December 6, 2012, the NRC accepted NEI 94-01, Revision 3-A. At that time, the NRC staff concluded NEI 94-01, Revision 3-A was acceptable for referencing in licensing applications for commercial nuclear power plants to the extent specified and under the limitations delineated in the NRC’s SE.

The NRC based its conclusion, in part, on the NEI 94-01, Revision 3-A Executive Summary which notes that the report meets the limitations and conditions of the SE for both Revision 2 and Revision 3. Revision 2-A of NEI 94-01 was issued in 2008 and included provisions for extending the ILRT interval to 15 years subject to the limitations and conditions provided in the SE for Revision 2. Revision 3-A was issued in July 2012 and included guidance for extending the Type C Local Leak Rate Testing (LLRT) interval to 75 months.

In letter dated August 20, 2013, the NRC requested that NEI 94-01, Revision 3, be updated. Revision 3 can be improved by stating that “TR NEI 94-01, Revision 3, as modified by conditions and limitations in Section 4.0 of the NRC SE for Revision 2 and in Section 4.0 of the NRC SE for Revision 3, is acceptable for referencing as the implementing document for meeting the performance-based requirements of 10 CFR 50, Appendix J - Option B.” This ANO-1 submittal will address the conditions and limitations presented in the SEs for both Revision 2 and Revision 3 of NEI 94-01.

In accordance with the guidance in NEI 94-01, Revision 2-A, ANO-1 proposes to extend the interval for the primary containment ILRT, which is currently required to be performed at ten year intervals to no longer than 15 years from the last ILRT. The current frequency would require the next ILRT to be performed during the spring 2015 refueling outage. The proposed amendment would allow the next ILRT for ANO-1 to be performed within 15 years from the last ILRT (i.e., December 16, 2005), as opposed to the current ten-year interval. This would allow

successive ILRTs to be performed at 15-year intervals (assuming acceptable performance history). The performance of fewer ILRTs will result in significant savings in radiation exposure to personnel, cost, and critical path time during future refueling outages.

2.0 PROPOSED CHANGE

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..

The proposed change would revise this portion of TS 5.5.16 by replacing the reference to RG 1.163 with a reference to NEI 94-01, Revision 3-A. The date for the next ILRT is also revised. The changes are underlined.

> 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 <u>NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012</u>, except that the next Type A test performed after the <u>December 16, 2005</u> Type A test shall be performed no later than <u>December 16, 2020</u>.

Attachment 2 contains the existing TS page 5.0-18 marked-up to show the proposed changes to TS 5.5.16.

3.0 BACKGROUND

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

In 1995, 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” was amended to provide a performance-based Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term “performance-based” in 10 CFR 50, Appendix J refers to both the performance history necessary to extend test intervals as well as to the criteria necessary to meet the requirements of Option B.

Also in 1995, RG 1.163 was issued. The RG endorsed NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J,” with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency from the containment Type A (ILRT) test from three tests in ten years to one test in ten years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, “Performance-Based Containment Leak-Test Program”, and Electric Power Research Institute (EPRI) TR-104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,” both of which illustrated that the risk increase associated with extending the ILRT surveillance interval was very small.

By letter dated April 11, 1996, Entergy Operations, Inc. (Entergy) submitted a TS change request concerning the implementation of 10 CFR 50, Appendix J, Option B. In the SE approving this request (letter dated October 3, 1996), the NRC noted the proposed TS changes were in compliance with the requirements of 10 CFR 50, Appendix J, Option B, and are consistent with the guidance in RG 1.163. Despite the different format of the ANO-2 TSs, all of the important elements of the guidance provided in the Staff’s letter to NEI dated November 2, 1995, are included in the proposed TS.

With the approval of the TS change request, ANO-2 transitioned to a performance-based ten-year frequency for the Type A tests.

Entergy submitted a TS change to extend the ILRT interval from ten years (120 months) to 15 years (180 months) via letter dated January 31, 2002. This one-time extension was approved by the NRC in letter dated September 24, 2002.

By letter dated August 31, 2007, NEI submitted Revision 2 of NEI 94-01 and EPRI TR-1009325, Revision 2, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,” to the NRC Staff for review.

NEI 94-01, Revision 2, describes an approach for implementing the optional performance-based requirements of Option B described in 10 CFR 50, Appendix J, which includes provisions for extending Type A intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. It delineates a performance-based approach for determining Type A, Type B, and Type C reactor building leakage rate surveillance testing frequencies. This method uses industry performance data, plant-specific performance data, and risk insights in determining the appropriate testing frequency. NEI 94-01, Revision 2, also discusses the performance factors that licensees must consider in determining test intervals. However, it does not address how to perform the tests because these details are included in existing documents (e.g., American National Standards Institute / American Nuclear Society [ANSI / ANS]-56.8-2002). The NRC

final SE issued by letter dated June 25, 2008, documents the NRC's evaluation and acceptance of NEI 94-01, Revision 2, subject to the specific limitations and conditions listed in Section 4.1 of the SE. The accepted version of NEI 94-01 was subsequently issued as Revision 2-A dated October 2008.

EPRI TR-1009325, Revision 2, provides a risk impact assessment for optimized ILRT intervals of up to 15 years, utilizing current industry performance data and risk-informed guidance, primarily Revision 1 of RG 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Bases." The NRC's final SE issued by letter dated June 25, 2008, documents the NRC's evaluation and acceptance of EPRI TR-1009325, Revision 2, subject to the specific limitations and conditions listed in Section 4.2 of the SE. An accepted version of EPRI TR-1009325 was subsequently issued as Revision 2-A (also identified as TR-1018243) dated October 2008.

By letter dated December 6, 2012, the NRC accepted NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance−Based Option Of 10 CFR Part 50, Appendix J". At that time, the NRC staff concluded NEI 94-01, Revision 3-A was acceptable for referencing in licensing applications for commercial nuclear power plants to the extent specified and under the limitations delineated in the NRC's safety evaluation (SE). Revision 3-A was issued in July 2012 and included guidance for extending the Type C Local Leak Rate Testing (LLRT) interval to 75 months.

4.0 TECHNICAL ANALYSIS

As required by 10 CFR 50.54(o), the ANO-1 reactor building is subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the ANO-1 10 CFR 50, Appendix J Testing Program Plan is based on RG 1.163, which endorses NEI 94-01, Revision 0. This license amendment request proposes to revise the ANO-1 10 CFR 50, Appendix J Testing Program Plan by implementing the guidance in NEI 94-01, Revision 3-A.

In the SE issued by the NRC dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE. The following addresses each of the six limitations and conditions associated with Revision 2 and the two limitations and conditions associated with Revision 3.

Limitation / Condition (from Section 4.1 of SE for Revision 2-A)	ANO-1 Response
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).	Following the NRC approval of this license amendment request, ANO-1 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 (see Attachment 7, "List of Regulatory Commitments").
2. The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests.	A schedule of containment inspections is provided in Section 4.3 below.
3. The licensee address the areas of the containment structure potentially subjected to degradation.	General visual examination of accessible interior and exterior surfaces of the containment system for structural problems is typically conducted in accordance with the ANO-1 Containment Inservice Inspection Plan which implements the requirements of the ASME, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g). The ANO-1 containment system does employ moisture barriers, but is not equipped with a sand cushion. There is one primary containment surface area that requires augmented examinations in accordance with ASME Section XI, IWE-1240. This is associated with the area around the equipment hatch.
4. The licensee addresses any test and inspections performed following major modifications to the containment structure, as applicable.	In December 2005, ANO-1 replaced the steam generators and the reactor vessel closure head that required modifications to the containment structure. A Type A test was performed post modification.
5. The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provisions of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition.	Entergy acknowledges and accepts this NRC staff position, as communicated to the nuclear industry in Regulatory Issue Summary (RIS) 2008-27, dated December 8, 2008.

Limitation / Condition (from Section 4.1 of SE for Revision 2-A)	ANO-1 Response
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 containment ILRT data.	Not applicable. ANO-1 is not licensed pursuant to 10 CFR Part 52.

Limitation / Condition (from Section 4.1 of SE for Revision 3-A)	ANO-1 Response
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 must be 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), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months. This is Topical Report Condition 1.	Following the NRC approval of this license amendment request, ANO-1 will provide the information as requested in this condition if a Type C frequency is extended (see Attachment 7, “List of Regulatory Commitments”).

Limitation / Condition (from Section 4.1 of SE for Revision 3-A)	ANO-1 Response
2. When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations. This is Topical Report Condition 2.	Following the NRC approval of this license amendment request, ANO-1 will provide the information as requested in this condition if a Type C frequency is extended (see Attachment 7, "List of Regulatory Commitments").

The current ANO-1 TS does not list any exceptions to the guidelines contained in RG 1.163.

4.1 Previous ILRT Results

A total of six Type A tests have been performed during the period of time of the ANO-1 Operating License. A summary of the last two ILRTs for ANO-1 is provided below.

April 16, 1992 – The as-left Type A test for containment leakage utilized the 95% upper confidence limit for leakage, water level corrections, and penalty for valves not being aligned in accident condition. The as-left leakage rate was calculated at 0.1245 wt%/day and 0.0841 wt%/day for Total Time and Mass Point calculational methods, respectively. Therefore containment leakage was less than 0.150 wt%/day and acceptable. ILRT test pressure was at 59.8 psig. In consideration of the performance-based criteria described in NEI 94-01, Revision 3-A, that provides the basis for determining whether historical Type A testing is acceptable, the April 16, 1992, Type A as-left test results of 0.1245 wt%/day and 0.0841 wt%/day for Total Time and Mass Point methods, respectively, are acceptable from a Performance-Based approach and ensures leakage integrity of the containment structure.

NEI 94-01, Revision 3-A, delineates in part (section 8.0, Testing Methodologies for Type A, B, and C tests) that for the purposes of determining an acceptable Type A test for operability considerations, the as-found overall integrated leakage rate shall be calculated by adding the positive difference between the as-found Minimum Path Leakage Rate (MNPLR) and the as-left MNPLR for each pathway tested and adjusted prior to the ILRT (savings); and the as-found MNPLR of all leakage paths isolated during the performance of the ILRT. NEI 94-01, Revision 3-A, further goes on to state in part in Section 8.0 (specifically addressed in the Note) that because of the performance based emphasis on Type A testing, the performance leakage rate does not use the savings value, but is calculated as the sum of the Type A upper confidence limit and as-left minimum pathway leakage rate for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position. In addition, leakage pathways that were

isolated during performance of the test because of excessive leakage must be factored into the performance determination. The performance criterion for Type A tests is a performance leak rate of less than 1.0 La.

The primary performance-based objective of the Type A test is not to quantify an overall containment system leakage rate, but to ensure continued leakage integrity of the containment structure. Type B and C testing assures that individual penetrations are essentially leak tight. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths. The performance criterion for Type A test allowable leakage is a performance leakage rate of less than 1.0 La. This allowable performance leakage rate is calculated as the sum of the Type A upper confidence limit and as-left MNPLR for all Type B and Type C pathways that were in service, isolated, or not aligned in the test position prior to performing the Type A test. In addition, leakage pathways that were isolated during performance of the ILRT because of excessive leakage must be factored into the performance determination.

December 16, 2005 – Type A test conducted at 58 psig test pressure was successfully performed (design pressure is 59 psig). An acceptable mass point leakage rate at the 95% upper confidence limit of 0.0643 wt%/day was calculated, not including the leakage rate corrections. Total leakage rate corrections (water volume corrections and Type B and C LLRT penalties for valves not aligned in accident condition) were determined to be 0.000169 wt%/day and 0.0051 wt%/day for water level and penalties, respectively. The ILRT occurred at the end of 1R19 outage. This allowed some penetration boundaries to be repaired or adjusted prior to the ILRT. The total leakage savings (positive difference between the as-found MNPLR and the as-left MNPLR) as a result of any repairs/adjustments was determined to be 2,823 standard cubic centimeters per minute (sccm) or the equivalent of 0.0017 wt%/day in penalties. In consideration of the aforementioned inputs, the as-found containment leakage rate was calculated at 0.0713 wt%/day. The acceptance criteria for containment as-found leakage is 0.75 La (allowable leakage), or 0.15 wt%/day. Therefore, the as-found ILRT results were acceptable.

The as-left leakage rate was calculated utilizing the 95% upper confidence limit of 0.0643 wt%/day, water level correction of 0.000169 wt%/day, and 0.0051 wt%/day penalty for valves not being aligned in accident alignment. Therefore, the as-left leakage rate was 0.0696 wt%/day. The acceptance criterion for the as-left ILRT is 0.15 wt%/day, so at 0.0696 wt%/day, the leakage results were acceptable.

Based on the above discussion, a test frequency of at least once per 15 years would be in accordance with NEI 94-01, Revision 3-A.

No modifications that require a Type A test are planned prior to 1R28, when the next Type A test will be performed under this proposed change. Any unplanned modifications to the reactor 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. There have been no pressure or temperature excursions in the reactor building which could have adversely affected reactor building integrity. There is no anticipated addition or removal of plant hardware within the reactor building which could affect leak-tightness.

4.2 Type B and Type C Testing Program

The ANO-1 Appendix J, Type B and Type C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the program as required by 10 CFR 50, Appendix J, Option B and TS 5.5.16. The 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.

A review of the most recent Type B and Type C test results and their comparison with the allowable leakage rate was performed. The combined Type B and Type C leakage acceptance criterion is 199,663 sccm. The maximum and minimum pathway leakage rate summary totals for the last two refueling outages are shown below:

1R22 As-Found Minimum Pathway Leakage	11,642 sccm
1R22 As-Left Maximum Pathway Leakage	26,193 sccm
1R23 As-Found Minimum Pathway Leakage	19,734 sccm
1R23 As-Left Maximum Pathway Leakage	35,601 sccm
1R24 As-Found Minimum Pathway Leakage	10,361 sccm
1R24 As-Left Maximum Pathway Leakage	19,599 sccm

As discussed in NUREG-1493 and NEI 94-01, Revision 3-A, Type B and Type C tests can identify the vast majority (greater than 95%) of all potential containment leakage paths. This amendment request 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. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

4.3 Supplemental Inspection Requirements

Attachment 4 provides a list of the summaries of the results from various ANO-1 reactor building inspections. It should be noted that in 1999 ANO committed to ASME IWL requirements. IWL requires that the site inspect both the tendons and the outside concrete. Prior to 1999, ANO was committed to RG 1.35 with respect to required tendon inspections.

Table 4-1 presents the results of the tendon surveillances. Note the 25-year, 30-year and 35-year surveillances are performed under IWL requirements hence the tendons and exterior concrete are both inspected. The ANO-1 IWL tendon inspection program conducts inspections every 5 years. The last inspection was performed during the last refueling outage (1R24)

Table 4-2 presents summaries of the results from the ANO-1 reactor building interior and exterior structural inspection surveillances. These surveillances were performed during each refueling shutdown and prior to any integrated leak test.

Table 4-3 presents the IWE inspection summary results. ANO has three inservice inspection (ISI) periods during each 10 year Interval. Initially ANO performed a reactor building dome and barrel IWE inspection. Subsequently, ANO 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 will perform both a barrel and dome ISI.

The current testing frequencies for Type B and C test are not affected by this requested amendment to permanently extend the Type A test interval to 15 years.

4.4 Deficiencies Identified

Consistent with the guidance provided in NEI 94-01, Revision 2, Section 9.2.3.3, abnormal degradation of the primary containment structure identified during the conduct of IWE / IWL program examinations or at other times is entered into the corrective action program for evaluation to determine the cause of the degradation and to initiate appropriate corrective actions. Attachment 5 presents a list of components that have failed the Type B or Type C tests from 2002 to present.

4.5 Plant-Specific Confirmatory Analysis

4.5.1 Methodology

An evaluation has been performed to assess the risk impact of extending the ANO-1 ILRT interval from the current ten years to 15 years. This plant-specific risk assessment followed 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 on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change the licensing basis of the plant. The current ANO-1 Level 1 and Large Early Release Frequency (LERF) internal events PRA model was used to perform the plant-specific risk assessment. This PRA model has been updated to meet Capability Category II of ASME PRA Standard RA-Sb-2005 and RG 1.200, Revision 1. The analyses include evaluation for the dominant external events (seismic and fire) using conservative expert judgment with the information from the ANO-1 Individual Plant Examination of External Events (IPEEE). Though the IPEEE seismic and fire event models have not been updated since the original IPEEE, the insights and information of IPEEE have been used to estimate the effect on total LERF of including these external events in the ILRT interval extension risk assessment.

In the SE issued by NRC letter dated June 25, 2008, the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE. The following table addresses each of the four limitations and conditions for the use of EPRI TR-1009325, Revision 2.

Limitation/Condition (From Section 4.2 of SE)	ANO-1 Response
1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension	ANO-1 PRA quality is addressed in Section 4.5.2.
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 3.2.4.5 of this SE. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is restrictive. In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in a previous one-time ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point.	EPRI Report No. 1009325, Revision 2-A, incorporates these population dose and Conditional Containment Failure Probability (CCFP) acceptance guidelines, and these guidelines have been used for the ANO-1 plant specific assessment.
3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate accident case (accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a	EPRI Report No. 1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (accident case 3b), and this value has been used in the ANO-1 plant specific risk assessment.
4. A licensee amendment request (LAR) is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance	ANO-1 does not rely on containment overpressure to assure adequate net positive suction head for ECCS pump following design basis accidents

4.5.2 PRA Quality

The ANO-1 PRA model is composed of a Level 1 (Revision 4p02) and LERF models for internal events. 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 ANO-1 PRA was initially developed in response to the NRC Generic Letter (GL) 88-20 for Individual Plant Examinations (IPE) which underwent NRC review. 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 model is maintained in accordance with Entergy PRA procedures.

The ANO-1 PRA internal events model has been upgraded to meet the RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1 standards. The industry peer review of the updated PRA model has been performed. The updated PRA model meets ASME Capability Category II requirements by addressing gaps identified by the peer review. As such, the updated ANO-1 PRA model is considered acceptable for use in assessing the risk impact of extending the ANO-1 reactor building ILRT interval to 15 years.

4.5.3 Summary of Plant-Specific Risk Assessment Results

The findings of the ANO-1 risk assessment confirm the general findings of previous studies that the risk impact associated with extending the ILRT interval from three in ten years to one in 15 years is small. The ANO-1 plant-specific results for extending ILRT interval from the current 10 years to 15 years are summarized below.

- The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current ten-year interval to a fifteen year interval is 1.70E-4 person-rem/year.
- The risk increase in LERF from extending the ILRT test frequency interval is 1.09E-8/yr.
- The change in CCFP from the current 10-year interval to a 15-year interval is 3.77E-3.
- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the risk impact on the total integrated plant risk by only 0.189%. Also, the change in Type A test frequency from the original three-per-ten-years to once-per-fifteen-years increases the risk only 0.455%. Therefore, the risk impact when compared to other severe accident risks is negligible.
- RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once per-fifteen-years is 1.09E-8/yr. Guidance in RG 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from 10 to 15 years is therefore considered non-risk significant and the results support this determination. In addition, the change in LERF resulting from a change in the Type A ILRT test interval from a three-per-ten-years to a once per-fifteen-years is 2.63E-8/yr, is also below the guidance threshold.

- RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in CCFP was estimated to be 3.77E-3 for the proposed change and 9.04E-3 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. These changes are small and demonstrate that the defense-in-depth philosophy is maintained.

In reviewing these results, the ANO-1 analysis demonstrates that the change in plant risk is small as a result of this proposed extension of ILRT testing. The change in LERF defined in the analysis for both the baseline and the current cases is within the acceptance criterion.

Details of the ANO-1 risk assessment are contained in Attachment 6 to this enclosure.

4.6 Conclusion

NEI 94-01, Revision 3-A, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. Entergy is adopting the guidance of NEI 94-01, Revision 3-A for the ANO-1 10 CFR Appendix J testing program plan.

Based on the previous ILRT tests conducted at ANO-1, it may be concluded that extension of the containment ILRT interval from 10 to 15 years represents minimal risk to increased leakage. Any risk is minimized by continued Type B and Type C testing performed in accordance with Option B of 10 CFR 50, Appendix J and inspection activities performed as part of the ANO-1 IWE / IWL ISI program.

This experience is supplemented by risk analysis studies, including the ANO-1 risk analysis provided in Attachment 6. The findings of the ANO-1 risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from ten to 15 years results in a very small change to the ANO-1 risk profile.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR 50, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants." Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing and reporting requirements for each type of test.

RG 1.163 was developed to endorse NEI 94-01, Revision 0 with certain modifications and additions.

The adoption of the Option B performance-based containment leakage rate testing for Type A testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, the test frequency is based upon an evaluation that review "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

NEI 94-01, Revision 3-A, describes an approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. The document incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 3-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate test frequencies. In the SE issued by NRC letter dated June 8, 2012, the NRC concluded that NEI 94-01, Revision 3, describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions, noted in Section 4.0 of the SE.

In letter dated August 20, 2013, the NRC requested that NEI 94-01, Revision 3, be updated. Revision 3 can be improved by stating that "TR NEI 94-01, Revision 3, as modified by conditions and limitations in Section 4.0 of the NRC SE for Revision 2 and in Section 4.0 of the NRC SE for Revision 3, is acceptable for referencing as the implementing document for meeting the performance-based requirements of 10 CFR 50, Appendix J - Option B."

EPRI TR-1009325, Revision 2, provides a risk impact assessment for optimized Integrated Leak Rate Test (ILRT) intervals up to 15 years, utilizing current industry performance data and risk-informed guidance. NEI 94-01, Revision 9-A, states that a plant-specific risk impact assessment should be performed using the approach and methodology described in

TR-1009325, Revision 2, for a proposed extension of the ILRT interval to 15 years. In the safety evaluation (SE) issued by NRC letter June 25, 2008, the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE.

Based on the considerations above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, Entergy Operations, Inc. (Entergy) has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements / criteria.

5.2 No Significant Hazards Consideration

A change is proposed to the Arkansas Nuclear One, Unit 1 (ANO-1), Technical Specifications 5.5.16, "Reactor Building Leakage Rate Testing Program." The proposed amendment would replace the reference to Regulatory Guide (RG) 1.163 with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, dated July 2012, as the implementation document used by ANO-2 to develop the ANO-2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. The proposed amendment would also extend the interval for the primary containment integrated leak rate test (ILRT), which is required to be performed by 10 CFR 50, Appendix J, from ten years to no longer than 15 years from the last ILRT.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment involves changes to the ANO-1 Reactor Building Leakage Rate Testing Program. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary reactor building function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor building itself and the testing requirements to periodically demonstrate the integrity of the reactor building exist to ensure the plant's ability to mitigate the consequences of an accident, do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

The integrity of the reactor building is subject to two types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and / or component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that the reactor building containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor building itself combined with the reactor building inspections performed in accordance with ASME, Section XI, the Maintenance Rule and regulatory commitments serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed amendment does not involve a significant increase in the consequences of an accident previously evaluate.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, for development of the ANO-1 performance-based testing program. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less the values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in RG 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. ANO-1 has determined that the increase in Conditional Containment Failure Probability due to the proposed change would be very small. Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, for the development of the ANO-1 performance-based leakage testing program, and establishes a 15-year interval for the performance of the reactor building ILRT. The reactor building and the testing requirements to periodically demonstrate the integrity of the reactor building exist to ensure the plant's ability to mitigate the consequences of an accident, do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, for the development of the ANO-1 performance-based leakage testing program, and establishes a 15 year interval for the performance of the containment ILRT. This amendment does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the Reactor Building Leakage Rate Testing Program, as defined in the TS, ensure that the degree of the reactor building structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall reactor building leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests will be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 3-A.

   Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current ANO-1 risk model concluded that extending the ILRT test interval from ten years to 15 years results in an acceptably small change to the ANO-1 risk profile.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

Similar TS changes were approved for ANO-2 on April 7, 2011 (ML110800034), Palisades on April 23, 2012 (ML120740081), and Nine Mile Point Unit 2 on March 30, 2010 (ML100730032).

Attachment 2 to

1CAN121302

Markups of Technical Specification Pages

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 ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995~~NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012, except that the next Type A test performed after the April 16, 1992 Type A test shall be performed no later than December 16, 2020 ~~April 15, 2007~~.

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 $\geq$ 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.

Attachment 3 to

1CAN121302

Clean (Revised) Technical Specification Pages

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 3-A, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J,” dated July 2012, except that the next Type A test performed after the April 16, 1992 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 $\geq$ 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.

Attachment 4 to

1CAN121302

Summary of the Results From Reactor Building Inspections

**Summary of the Results From
Reactor Building Inspections**

Below is a list of ANO-1 reactor building inspections and summaries of the results of those inspections. Note in 1999 ANO committed to IWL requirements. IWL requires that the site inspect both the tendons and the outside concrete. Prior to 1999 ANO was committed to Regulatory Guide 1.35 with required tendon inspections.

Table 4-1 is for the results of the tendon surveillances. Note the 25-year, 30-year and 35-year surveillances are performed under IWL requirements hence the tendons and exterior concrete are both inspected.

Table 4-2 presents summaries of the results from the ANO-1 reactor building interior and exterior structural inspection surveillances. These surveillances were performed during each refueling shutdown and prior to any integrated leak test.

Table 4-3 presents the IWE inspection summary results. ANO has three ISI periods during each 10 year Interval. Initially ANO performed a reactor building dome and barrel IWE inspection. Subsequently, ANO 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 will perform both a barrel and dome ISI inspection.

Table 4-1
Tendon Surveillances

May 29, 1974	First inspection for the reactor building tendon end anchorage concrete surveillance program. The results of the inspection were that "to this date no detrimental concrete cracking has occurred in the vicinity of the tendon end anchorages."
October 1977	This report is for the three year reactor building tendon surveillance. The report concluded that "Based on the results of the three-year tendon surveillance test reported herein, the conclusion is reached that no abnormal degradation of the containment structure is indicated for Arkansas Nuclear One Unit No. 1."
October 15, 1979	Revised sheets for the five-year reactor building tendon surveillance report were submitted to the NRC. The report concluded that "Based on the results of the five-year tendon surveillance test reported herein, the conclusion is reached that no abnormal degradation of the containment structure post-tensioning system is indicated for Arkansas Nuclear One – Unit 1."
August 1, 1983	Inspection report for the ten year physical tendon surveillance of Arkansas Nuclear One – Unit 1. The report concluded that "Based on the results of this surveillance, no abnormal degradation of the containment post tensioning is evident. "
July 20, 1988	15 year tendon surveillance report. The report concluded that "Based upon the results of the Fifteenth Year Physical Surveillance of the Unit 1 containment building at Arkansas Nuclear One reported herein, the conclusion is reached that no abnormal degradation has occurred in the post-tensioning system."
March 1, 1994	20th year physical surveillance of the ANO-1 reactor building post-tensioning system was performed. "The results of the surveillance conclude that no abnormal degradation has occurred in the post tensioning system."
June 8, 2001	25 year tendon surveillance and concrete surface inspection report. The 25th year tendon surveillance and concrete surface inspection was completed in 1999. The report concluded that "Based upon an evaluation of the In-Service Inspection results addressed in the report, the ANO Unit 1 Reactor Building structure has experienced no abnormal degradation of the building nor the post tensioning system."

December 7, 2004	30th year tendon surveillance and concrete inspection. The report concluded that "Entergy concludes that the ANO Unit 1 containment is currently capable of performing its design function and should remain capable of performing its design function until completion of the 35 year tendon surveillance and concrete inspection. "
May 18, 2009	35th year tendon inspection and concrete inspection. This is the ASME Code IWL examination. The inspection report concluded that "Based upon the evaluation of the In-Service Inspection results for the 35th Year Containment IWL Inspection of the Arkansas Nuclear One Unit 1 Containment Structure Post Tensioning System, reported herein, PSC concludes that the containment structure has experienced no abnormal degradation of the post-tensioning system. The containment post tensioning system is performing in accordance with the design requirements and is expected to continue to do so for the life of the unit."
May 2013	40th year IWL inspection and concrete inspection.. Based on the results of the 40th year tendon surveillance and concrete inspection, Entergy concludes that the ANO Unit 1 reactor building is currently capable of performing its design function and should remain capable of performing its design function until completion of the 45th year tendon surveillance and concrete inspection. Additionally, the observed indications do not indicate the presence of degradation in inaccessible areas.

Table 4-2
Reactor Building Interior and Exterior Structural Inspections

November 21, 1986	No concerns were noted that degrade the structural integrity and leak tightness of the Unit 1 reactor building.
October 13, 1988	The hair line cracks noted do not affect the structural integrity of the Unit 1 reactor building.
December 11, 1990	No deficiencies were noted.
April 8, 1992	No cracks, separations, or other surface defects were observed which would affect the reactor building structural integrity and leak tightness.
October 6, 1993	Deficiencies noted were judged to no affect structural integrity.
December 22, 1995	No deficiencies were noted.
September 23, 1996	No deficiencies were noted that would compromise reactor building integrity.
May 1, 1998	Numerous deficiencies noted during the inspection. All were evaluated and it was determined that none of the deficiencies would affect reactor building integrity.
April 22, 2000	Numerous deficiencies noted during the inspection. All were evaluated and it was determined that none of the deficiencies would affect reactor building integrity.
April 5,, 2001	Numerous deficiencies noted during the inspection. All were evaluated and it was determined that none of the deficiencies would affect reactor building integrity.
December 14, 2005	All deficiencies noted were judged to be non-significant.
November 3, 2008	Several deficiencies were noted during the inspection. All were evaluated and it was determined that none of the deficiencies would affect reactor building integrity.
March 2013	Several deficiencies were noted during the inspection. All were evaluated and it was determined that none of the deficiencies would affect reactor building integrity.

Table 4-3
IWE Inspections

March 2001	Barrel liner plates (partial inspection, 209 out of 215 totals, most barrel liner plates were inspected however some were missed. The missed ones were inspected in the May 2004 outage), ceiling transition liner plates, upper dome liner plates and lower dome liner plates were all inspected. Inspected the moisture barrier between the liner plate and the concrete surfaces. Numerous deficiencies (blisters, rust, flaking, mechanical damage, arc strikes, and bulging) were noted during these inspections. Evaluations concluded that none of the deficiencies affect containment integrity.
October 2002	No IWE inspections were scheduled. The only IWE activity involved the identification of a bubble in the reactor building top coat. The area prepped, primed and painted.
May 2004	Visual inspections were performed on the Equipment Hatch, Escape Hatch, and Personnel Hatch bolting. Also, a visual inspection was performed on the Equipment Hatch. Visual inspections were performed on 9 barrel liner plates – several of which were missed in the March 2001 inspection. None of the noted deficiencies found during these inspections affect containment building integrity.
October 2005	Visual inspections were performed on the reactor building dome. The dome includes the transition liner plates between the barrel and dome, dome center liner plate, dome vent liner plates, lower dome liner plates, and the upper dome liner plates. In addition the moisture barrier between the liner plate and concrete surfaces was inspected. Also the Equipment Hatch was inspected along with its bolting and the Personnel Hatch bolting. Several deficiencies were noted. The inspection report concluded that none of the deficiencies affected the liner plate structural integrity.
May 2007	Visual inspections and ultrasonic inspections were performed on the reactor building barrel liner plates. The surface areas of the Personnel and Escape hatch were also inspected. Numerous deficiencies (blistering, rust, flaking, budging, mechanical damage, etc.) were noted during the liner plate inspections. It was concluded that the noted deficiencies did not affect the liner plate structural integrity.
Nov. 2008	Visual inspection was performed on the reactor building dome. In addition the moisture barrier between the steel liner plate and the concrete surfaces was inspected. Also, the Personnel Hatch and Escape Hatch bolted connections were inspected. Several deficiencies were noted. The inspection report concluded that none of the deficiencies affected the liner plate structural integrity.

April 2010	Visual inspections and ultrasonic inspections were performed on the reactor building barrel liner plates. The surface areas of the Personnel and Escape Hatch were also inspected. Numerous deficiencies (blistering, rust, flaking, budging, mechanical damage, etc.) were noted during the liner plate inspections. It was concluded that the noted deficiencies did not affected the liner plate structural integrity.
October 2011	Visual inspections were performed on the reactor building dome liner plates. In addition, IWE inspections were performed on the Escape Hatch bolted connections, the moisture barrier between the steel liner and concrete surfaces, and the reactor building sump. Per the general visual examination report, “No new areas were identified during this examination; all identified conditions are visually the same as observed in 1R19.” In summary, no identified issues would affect reactor building integrity.
May 2013	Visual and ultrasonic inspections were performed on the reactor building barrel liner plates. The surface areas of the Personnel and Escape Hatch were also inspected. Numerous deficiencies (blistering, rust, flaking, budging, mechanical damage, etc.) were noted during the liner plate inspections. It was concluded that the noted deficiencies did not affected the liner plate structural integrity.

Attachment 5 to

1CAN121302

List of Components that Failed Type B or Type C Tests Since 2002

List of Components that Failed Type B or Type C Tests Since 2002

Below is a list of ANO-1 components that have failed a LLRT's administrative limit in either the Operations procedure or Engineering procedure from 2002 to present.

SV-1818/PSV-1800	October 8, 2002 (2500 sccm)
SV-1818/PSV-1800	October 19, 2002 (4500 sccm)
SV-1818/PSV-1800	May 6, 2004 (4800 sccm)
SV-1818/PSV-1800	October 14, 2005 (3400 sccm)
SV-1818/PSV-1800	April 28, 2007 (5800 sccm)
SV-1818/PSV-1800	March 24, 2010 (4850 sccm)
SV-1818/PSV-1800	October 20, 2011 (10500 sccm)
SV-1818/PSV-1800	November 15, 2011 (3500 sccm)
SV-1818/PSV-1800	May 3, 2013 (5900 sccm)
SV-1840	October 8, 2002 (5900 sccm)
CV-4804	April 26, 2004 (1460 sccm)
CV-4804	October 13, 2005 (1340 sccm)
CV-4804	November 2, 2008 (1800 sccm)
CV-4804	March 23, 2010 (2010 sccm)
CV-4804	March 27, 2013 (46100 sccm)
MU-36B	October 23, 2011 (11500 sccm)
HV-150	October 11,, 2002 (13200 sccm)
HV-150	May 1, 2004 (5000 sccm)
HV-150	October 12, 2005 (3000 sccm)
HV-150	April 5, 2010 (3050 sccm)
HV-150	April 8, 2010 (3700 sccm)
HV-150	November 1, 2011 (5100 sccm)
CV-6205	April 10, 2013 (74800 sccm)
SV-1440	October 22, 2011 (3300 sccm)
SV-1440	October 22, 2011 (1660 sccm)
Escape Hatch Outer Door Seal	November 16, 2011 (2483 sccm)
Personnel Hatch Outer Door Seal	November 17, 2011 (560 sccm)
Personnel Hatch Inner Door Seal	March 20, 2010 (2940 sccm)
Personnel Hatch Inner Door Seal	October 11, 2011 (550 sccm)
Personnel Hatch Inner Door Seal	November 17, 2011 (1310 sccm)
Personnel Hatch Inner Door Seal	April 21, 2010 (930 sccm)

Attachment 6 to

1CAN121302

Risk Analysis

Table of Contents

List of Tables

1.0 PURPOSE

The purpose of this report is to provide an estimation of the change in risk associated with extending the Type A integrated leak rate test interval beyond the current 10 years specified by 10 CFR 50, Appendix J, Option B for Arkansas Nuclear One Unit 1 (ANO-1)[1]. This activity supports a request for an exemption from the performance of the integrated leak rate test (ILRT) during the planned 1R25 outage. The assessment is consistent with the processes described in the methodology identified in EPRI's guidance document, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals[2].

1.1 SUMMARY OF THE ANLYSIS

10 CFR 50, Appendix J allows individual plants to extend Type A surveillance testing requirements and to provide for performance-based leak testing. This report documents a risk-based evaluation of the proposed change of the ILRT interval for the ANO-1. The proposed change would impact testing associated with the current surveillance tests for Type A leakage, procedure 5120.400[3]. No change to Type B or Type C testing is proposed at this time.

This analysis utilizes the guidelines set forth in NEI 94-01[4], the methodology used in the EPRI Report and considers the submittals generated by other utilities.

This calculation evaluates the risk associated with various ILRT intervals as follows:

- 3 years – Interval based on the original requirements of 3 tests per 10 years.
- 10 years – This is the current test interval required for ANO-1.
- 15 years – Proposed extended test interval.

The analysis utilizes the ANO-1 PRA results taken from the LERF (large early release frequency) model[5].

The release category and person-rem information is based on the approach suggested by EPRI's guidance document.

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1 Appendix J to Part 50 – Primary Reactor Containment Leakage Testing for Water-Cooling Power Reactors, U.S. Nuclear Regulatory Commission (USNRC), 10 CFR Part 50, Appendix J, January 2006.
2 Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.
3 Unit One Integrated Leak Rate Test, Entergy Operations Incorporated, 5120.400, December 17, 2005.
4 Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 0, Nuclear Energy Institute, NEI 94-01, July 26, 1995.
5 ANO-1 Large Early Release Frequency (LERF) Model, Rev. 1, Entergy Operations Incorporated, PRA-A2-01-001S12, June 2009.

1.2 SUMMARY OF RESULTS/CONCLUSIONS

The specific results are summarized in Table 1 below. The Type A contribution to LERF is defined as the contribution from Class 3b. A breakdown of all the EPRI classifications is contained in Table 9 and 10 of this report.

Table 1
Summary of Risk Impact on Extending Type A ILRT Test Frequency

	Risk Impact for 3-years (baseline)	Risk Impact for 10-years (current requirement)	Risk Impact for 15-years
Total integrated risk (person-rem/yr)	8.97E-2	8.99E-2	9.01E-2
Type A testing risk (person-rem/yr)	1.06E-4	3.53E-4	5.29E-4
% total risk (Type A / total)	0.118%	0.392%	0.587%
Type A LERF (Class 3b) (per year)	6.56E-9/yr	2.19E-8/yr	3.28E-8/yr
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			1.70E-4
% Increase from current (Δ Risk / Total Risk)			0.189%
Δ LERF from current (per year)			1.09E-8
Δ CCFP from current			3.77E-3
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			4.08E-4
% Increase from baseline (Δ Risk / Total Risk)			0.455%
Δ LERF from baseline (per year)			2.63E-8
Δ CCFP from baseline			9.04E-3

The results are discussed below:

- The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current ten-year interval to a fifteen year interval is 1.70E-4 person-rem/year.

- The risk increase in LERF from extending the ILRT test frequency from the current 10-year interval to a 15-year interval is 1.09E-8/yr.

- The change in conditional containment failure probability (CCFP) from the current 10-year interval to a 15-year interval is 3.77E-3.

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the risk impact on the total integrated plant risk by only 0.189%. Also, the change in Type A test frequency from the original three-per-ten-years to once-per-fifteen-years increases the risk only 0.455%. Therefore, the risk impact when compared to other severe accident risks is negligible.

- Reg. Guide 1.174[6] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide (RG) 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once per-fifteen-years is 1.09E-8/yr. Guidance in RG 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from 10 to 15 years is therefore considered non-risk significant and the results support this determination. In addition, the change in LERF resulting from a change in the Type A ILRT test interval from a three-per-ten-years to a once per-fifteen-years is 2.63E-8/yr, is also below the guidance.

- RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 3.77E-3 for the proposed change and 9.04E-3 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. These changes are small and demonstrate that the defense-in-depth philosophy is maintained.

In reviewing these results, the ANO-1 analysis demonstrates that the change in plant risk is small as a result of this proposed extension of ILRT testing. The change in LERF defined in the analysis for both the baseline and the current cases is within the acceptance criterion.

In addition to the baseline assessment, three sensitivity exercises are included. These analyses are provided in Section 5 and are consistent with those outlined in the EPRI guidance document[7].

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6 An Approach for Using Probabilistic Risk Assessment in Risk-Informed decisions on Plant-Specific Changes to the Licensing Basis, U.S. Nuclear Regulatory Commission (USNRC), Regulatory Guide 1.174, July 1998.

7 Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

2.0 DESIGN INPUTS

The ANO-1 PRA is intended to provide “best estimate” results that can be used as input when making risk informed decisions. The PRA provides the most recent results for the ANO-1 PRA. The inputs for this calculation come from the information documented in the ANO-1 PRA Large Early Release Model[8]. The ANO-1 release states are summarized in Table 2. ANO-1 Level 2 results are lumped into 4 sequence states that represent the summation of individual accident categories. The number of sequences comprising each sequence state is also presented in Table 2.

Table 2
Release Category Frequencies

Release Category	Contributing ANO-1 Accident Categories	Frequency (/yr)	EPRI Classification
INTACT (S)[1]	10	2.65E-6	Class 1
LERF	18	5.82E-8	Class 8
SERF	9	4.97E-11	Class 6
LATE	14	1.97E-7	Class 7
Total	n/a	2.90E-6	n/a

The LERF contribution for ANO-1 contains early containment failures due to containment phenomenon and by the EPRI guidance should be collected in Class 7. To accurately classify the contributions, the LERF contribution is separated to be consistent with the guidance document.

Table 4.3-2 of ANO-1 LERF model analysis provides the endstate and frequency of the respective endstate. Table 3 shows the classification of each endstate and the totals of each classification. The description of the outcome is used to classify each of the 18 contributing endstates.

8 ANO-1 Large Early Release Frequency (LERF) Model, Rev. 1, Entergy Operations Incorporated, PRA-A2-01-001S12, June 2009.

Table 3
Decomposition of ANO-1 LERF Frequency and EPRI Classification

Endstate	Description of Outcome	Frequency (per year)	EPRI Class
LERF01	Containment Failure following high-pressure (HP) vessel breach (VB) – Non-SBO	8.48E-11	7
LERF02	Containment Failure following HP VB – Non-SBO	6.17E-10	7
LERF03	Containment Failure following Low Pressure (LP) VB – Non-SBO	1.10E-10	7
LERF04	Temperature Induced (TI) SGTR – Non-SBO	1.51E-8	8
LERF05	Containment Failure following LP VB – Non-SBO	2.25E-10	7
LERF06	Pressure Induced (PI) SGTR – Non-SBO	3.42E-9	8
LERF07	Containment Failure following LP VB – Non-SBO	1.39E-8	7
LERF08	Loss of Isolation – Non-SBO	2.87E-9	2
LERF09	Containment bypass – Non-SBO	1.71E-8	8
LERF10	Containment Failure following LP VB - SBO	0.00E+0	7
LERF11	Containment Failure following HP VB - SBO	4.46E-10	7
LERF12	Containment Failure following LP VB - SBO	9.09E-11	7
LERF13	TI-SGTR – SBO	6.81E-11	8
LERF14	Containment Failure following LP VB – SBO	1.19E-10	7
LERF15	PI-SGTR – SBO	2.87E-9	8
LERF16	Containment Failure following LP VB – SBO	0.00E+0	7
LERF17	Loss of Isolation – SBO	6.50E-11	2
LERF18	Containment bypass – SBO	1.25E-9	8
Contribution to EPRI Classification 2		2.94E-9 /yr	
Contribution to EPRI Classification 7		1.56E-8 /yr	
Contribution to EPRI Classification 8		3.97E-8 /yr	
Total LERF		5.82E-8 /yr	

In order to develop the person-rem dose associated to the plant damage state it is necessary to associate each release category with an associated release of radionuclides and from this information to calculate the associated dose.

The EPRI guidance on leak rate testing[9] indicates that a surrogate can be applied and is acceptable for estimating risk and suggests one surrogate source is the results contained in NUREG 1150[10]. NUREG 1150 examined both pressurized water reactors (PWRs) and boiling water reactors (BWRs). The results presented for boiling water reactors (i.e, Peach Bottom, Grand Gulf) are not considered appropriate for this analysis since the core melt mechanics and design are substantially different between ANO-1 and the BWRs. Therefore, their results are excluded from consideration.

NUREG 1150 also analyzed Zion, Sequoyah and Surry PWR designs. Sequoyah utilizes an ice condenser design and the presence of ice and restricted flow paths can lead to sequences and conditions that are not found in a large dry containment design such as ANO-1. Therefore, Sequoyah is not considered a good PWR design for comparison.

Zion is a 4-loop Westinghouse design large dry containment and may be somewhat closer to the ANO-1 design. However the 4-loop design and power level may influence timing source term. Therefore it is not selected as a surrogate.

The remaining assessed design is Surry. It is a Westinghouse 3 loop design and given the power level and other factors, is considered the best surrogate after examination of the NUREG 1150 analyzed plants.

NUREG/CR-4551[11] provides the Level 2 analysis and offsite consequence assessment for Surry. Table 4.3-1 of that document provides a summary of consequence results that includes population dose (exposure) within 50 miles for internal events. A range of outcomes exists for each source term group based on the consequence measures. A matrix is formed and values provided for figures of merit.

The exposure estimates for a range of 50 miles around the Surry site are provided in Table 4 for each reported source term group.

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[9] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

[10] Reactor Risk Reference Document, Appendices J-O, Draft for Comment, United States Nuclear Regulatory Commission (USNRC), NUREG-1150, January 1987.

[11] Breeding, R. J., et al, Evaluation of Severe Accident Risks: Surry Unit 1, Main Report, Rev. 1, U.S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 3, October 1990.

Table 4
Reported Person Rem Estimates for Surry Source Term Groups
(summarized from NUREG/CR-4551)

Source Term Grouping	Outcome 1 (Sv[1])	Outcome 2 (Sv)	Outcome 3 (Sv)
SUR-01	NA	2.33E+3	1.25E+3
SUR-02	5.33E+3	1.13E+4	5.82E+3
SUR-03	1.15E+4	2.26E+4	1.13E+4
SUR-04	1.04E+4	1.45E+4	NA
SUR-05	NA	5.15E+4	2.62E+4
SUR-06	NA	2.42E+4	2.15E+4
SUR-07	2.76E+4	3.43E+4	1.46E+4
SUR-08	1.68E+4	2.14E+4	1.61E+4
SUR-09	1.36E+4	1.74E+4	NA
SUR-10	4.73E+4	4.66E+4	3.34E+4
SUR-11	4.56E+4	2.77E+4	2.78E+4
SUR-12	2.69E+4	3.01E+4	2.67E+4
SUR-13	2.15E+4	2.68E+4	NA
SUR-14	1.88E+4	2.23E+4	NA
SUR-15	4.28E-1	3.10E+0	NA
SUR-16	4.28E+0	3.75E+1	NA
SUR-17	2.66E+3	6.71E+3	NA
SUR-18	0.00E+0	NA	NA

1. Values provided in Sieverts (Sv). Conversion factor 1 Sv = 100 rem.

In order to utilize this information it is necessary to convert it to the form needed in the ILRT analysis. This involves classification into one of the three EPRI classes and then determining the representative person-rem estimates.

NUREG/CR-4551[12] provides some guidance with respect to the composition of the source term grouping. For example SUR-01 is dominated by bypass sequences. Using this information the Surry results are grouped to the EPRI classes. The grouping is presented in Table 5.

Table 5
Assignment of Surry Source Term Groups to EPRI Classes

EPRI Class	Surry Source Term Groups Applied[1]
Class 2	SUR-14
Class 7	SUR-04, SUR-07, SUR-08, SUR-09, SUR-11, SUR-12, SUR-13, SUR-15, SUR-16, SUR-17
Class 8	SUR-01, SUR-02, SUR-03, SUR-05, SUR-06, SUR-10

1. Group SUR-18 is not applied to an EPRI class since the listed outcomes in Table 3 are either 0.0 or NA.

The source term exposure estimates for each source term group are first averaged to obtain a value for the source term group and then the individual groups are averaged to obtain a class estimate. An example calculation is provided below.

Source term group (STG) SUR-01 has two estimates for exposure (see Table 4). These values are first averaged to obtain a STG average for SUR-1.

$$Sv_{avg} = (2.33E+3 + 1.25E+3)\ Sv\ /2 = 1.79E+3\ Sv \qquad \text{(eq. 1)}$$

Repeating this process arrives at the data provided in Table 6.

It is noted that for Class 7 and Class 8 there are multiple source term groups included. In these cases the individual results using Equation 1 for each contributing Surry STG were summed and then averaged to obtain an estimate for the EPRI class.

[12] Breeding, R. J., et al, Evaluation of Severe Accident Risks: Surry Unit 1, Main Report, Rev. 1, U.S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 3, October 1990.

Table 6
Average Person-Rem for Surry Source Term Groups

Source Term Group	Exposure (Sv)
SUR-01	1.79E+3
SUR-02	7.48E+3
SUR-03	1.51E+4
SUR-04	1.25E+4
SUR-05	3.89E+4
SUR-06	2.29E+4
SUR-07	2.55E+4
SUR-08	1.81E+4
SUR-09	1.55E+4
SUR-10	4.24E+4
SUR-11	3.37E+4
SUR-12	2.79E+4
SUR-13	2.42E+4
SUR-14	2.06E+4
SUR-15	1.76E+0
SUR-16	2.09E+1
SUR-17	4.69E+3
SUR-18	NA

These results are then grouped into the EPRI Classes using Table 5 and the average, minimum and maximum exposures are defined. The results are presented in Table 7 in units of person-rem.

Table 7
Average Person-Rem for EPRI Classes Based on Surry Source Term Groups

EPRI Class	Weighted Average Exposure (person-rem)	Max Exposure in Class (person-rem)	Min Exposure in Class (person-rem)
Class 1	5.76E+2	NA[1]	NA
Class 2	2.06E+6	NA[2]	NA
Class 7	1.62E+6	3.37E+6	1.76E+2
Class 8	2.14E+6	4.24E+6	1.79E+5

1. Intact containment dose rate from the EPRI Report.
2. Only one source term group applied.

EPRI's ILRT guidance document[13] utilizes a multiplication factor to develop the design basis leakage value (L_a) that is based on generic information that provides comparative local population ratios. The ANO-1 population dose is adjusted for the local plant-specific population using a "population dose factor". The population dose factor is used to adjust the Surry population dose to account for differences in the local populations of Surry and ANO-1. The population dose factor is calculated by dividing the ANO-1 population[14] by the Surry population information taken from EPRI's ILRT report.

Total ANO-1 Population = 250,860

Surry Population = 1,230,000

Population Dose Factor = 0.20

The relationship above implies that the resultant doses are a direct function of population within 50 miles of each site. This does not take into account differences in meteorology, environmental factors, containment designs or other factors but does provide a reasonable first-order approximation of the population dose as would be generated by the Surry sequences. The release category dose information is presented in Table 8.

[13] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

[14] Arkansas Nuclear One Emergency Plan: Revision 37, Entergy Operations Incorporated, February 2013

Table 8
ANO-1 Dose for EPRI Accident Classes

Release Category	Frequency (/yr)	EPRI Class	ANO-1 Dose (person-rem)
INTACT	2.65E-6	Class 1	1.15E+2
LERF[1]	2.94E-9	Class 2	4.12E+5
SERF[2]	4.97E-11	Class 6	4.12E+4[3]
LERF + LATE[4]	2.12E-7	Class 7	2.52E+5
LERF[5]	3.97E-8	Class 8	8.72E+5

1. The EPRI Class 2 category consists of ANO-1 assigned LERF contribution associated with isolation failures as re-categorized in Table 3.
2. The EPRI Class 6 category consists of ANO-1 assigned scrubbed isolation failures in SERF.
3. The EPRI Class 6 Does rate is derived from the Class 2 does rate. A decontamination factor of 10 is applied with the assumption that 10 percent of the release would be scrubbed.
4. The EPRI Class 7 category consists of the ANO-1 assigned LERF contribution associated with phenomenological failures as re-categorized in Table 3. Additionally consistent with the EPRI guidance document, LATE failures are classified as Class 7.
5. The EPRI Class 8 category consists of the ANO-1 assigned LERF contribution associated with bypass or SGTR failures as re-categorized in Table 3.

3.0 ASSUMPTIONS

1. The maximum containment leakage for EPRI Class 1 sequences is 1 La (Type A acceptable leakage) because a new Class 3 has been added to account for increased leakage due to Type A inspections[15].

2. The maximum containment leakage for Class 3a sequences is 10 La based on the EPRI guidance.

3. The maximum containment leakage for Class 3b sequences is 100 La based on the NEI guidance contained within the EPRI report.

4. Class 3b is conservatively categorized as LERF based on the NEI guidance and previously approved EPRI methodology.

5. Containment leakage due to EPRI Classes 4 and 5 are considered negligible based on the NEI guidance and the previously approved EPRI methodology.

[15] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008

6. The containment releases are not impacted with time.

7. The containment releases for EPRI Classes 2, 6, 7 and 8 are not impacted by the ILRT Type A Test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A.

8. Because EPRI Class 8 sequences are containment bypass sequences, potential releases are directly to the environment. Therefore, the containment structure will not impact the release magnitude.

4.0 CALCULATIONS

This calculation applies the ANO-1 PRA release category information in terms of frequency and person-rem estimates to determine the changes in risk due to increasing the ILRT test interval. The changes in risk are assessed consistent with the guidance provided in the EPRI guidance document.

The detailed calculations performed to support this report were of a level of mathematical significance necessary to calculate the results recorded. However, the tables and illustrational calculation steps presented may present rounded values to support readability.

4.1 CALCULATIONAL STEPS

The analysis employs the steps provided in EPRI's ILRT guidance document and uses associated risk metrics to evaluate the impact of a proposed change on plant risk. These measures are the change in release frequency, the change in risk as defined by the change in person-rem, the change in LERF, and the change in the conditional containment failure probability (CCFP).

Additionally EPRI also lists the change in core damage frequency as a measure to be considered[16]. Since the testing addresses the ability of the containment to maintain its function, the proposed change has no measurable impact on core damage frequency. Therefore, this attribute remains constant and has no risk significance.

The overall analysis process is documented as outlined below:

- Define and quantify the baseline plant damage classes and person-rem estimates.
- Calculate baseline leakage rates and estimate probability to define the analysis baseline.
- Develop baseline population dose (person-rem) and population dose rate (person-rem/yr).
- Modify Type A leakage estimate to address extension of the Type A test frequency and calculate new population dose rates, LERF and conditional containment failure probability.
- Compare analysis metrics to estimate the impact and significance of the increase related to those metrics.

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[16] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008

The first step in the analysis is to define the baseline plant damage classes and person-rem dose measures. Plant damage state information is developed using the ANO-1 PRA Level 2 PRA results[17]. The containment end state information and the results of the containment analysis are used to define the representative sequences. The population person-rem dose estimates for the key plant damage classes are based on the application of the method described in EPRI's ILRT guidance document.

The product of the person-rem for the plant damage classes and the frequency of the plant damage state is used to estimate the annual person-rem for the plant damage state. Summing these estimates produces the annual person-rem dose based on the sequences defined in the PRA.

The PRA plant damage state definitions considered isolation failures due to Type B and Type C faults and examined containment challenges occurring after core damage and/or reactor vessel failure. These sequences are grouped into key plant damage classes. Using the plant damage state information, bypass, isolation failures and phenomena-related containment failures are identified. Once identified, the sequence was then classified by the EPRI release category definitions. With this information developed, the PRA baseline inputs are completed.

The second step expands the baseline model to address Type A leakage. The PRA did address Type A (liner-related) faults and this contribution has been binned into EPRI Class 1. A new estimation using the EPRI methodology must be incorporated to provide a complete baseline. In order to define leakage that can be linked directly to the Type A testing, it is important that only failures that would be identified by Type A testing exclusively be included.

EPRI's ILRT guidance document[18] provides the estimate for the probability of a leakage contribution that could only be identified by Type A testing based on industry experience. This probability is then used to adjust the intact containment category of the ANO-1 PRA to develop a baseline model including Type A faults.

The release, in terms of person-rem, is developed based on information contained in EPRI's report and is estimated as a leakage increase relative to allowable dose (L_a) defined as part of the ILRT.

The predicted probability of Type A leakage is then modified to address the expanded time between testing. This is accomplished by a ratio of the existing testing interval and the proposed test interval. This assumes a constant failure rate and that the failures are randomly dispersed during the interval between the test.

The change due to the expanded interval is calculated and reported in terms of the change in release due to the expanded testing interval, the change in the population person-rem and the change in large early release frequency. The change in the conditional containment failure probability is also developed. From these comparisons, a conclusion is drawn as to the risk significance of the proposed change.

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[17] ANO-1 Large Early Release Frequency (LERF) Model, Rev. 1, Entergy Operations Incorporated, PRA-A2-01-001S12, June 2009.

[18] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008

Using this process, the following were performed:

1. Map the ANO-1 release categories into the 8 release classes defined by the EPRI Report.
2. Calculate the Type A leakage estimate to define the analysis baseline.
3. Calculate the Type A leakage estimate to address the current testing frequency.
4. Modify the Type A leakage estimates to address extension of the Type A test interval.
5. Calculate increase in risk due to extending Type A testing intervals.
6. Estimate the change in LERF due to the Type A testing.
7. Estimate the change in CCFP due to the Type A testing.

4.2 SUPPORTING CALCULATIONS

Step 1: Map the release categories into the 8 release classes defined by the EPRI Report[19]

EPRI defines eight (8) release classes as presented in Table 9.

Table 9
Containment Failure Classifications

Failure Classification	Description	Interpretation for Assigning ANO-1 Release Category
1	Containment remains intact with containment initially isolated	Intact containment bins or late basemat attack sequences.
2	Dependent failure modes or common cause failures	Isolation faults that are related to a loss of power or other isolation failure mode that is not a direct failure of an isolation component
3	Independent containment isolation failures due to Type A related failures	Isolation failures identified by Type A testing
4	Independent containment isolation failures due to Type B related failures	Isolation failures identified by Type B testing
5	Independent containment isolation failures due to Type C related failures	Isolation failures identified by Type C testing
6	Other penetration failures	Isolation failure with scrubbing or small isolation fails
7	Induced by severe accident phenomena	Early containment failure sequences as a result of hydrogen burn or other early phenomena
8	Bypass	Bypass sequence or SGTR

Table 10 presents the ANO-1 release category mapping for these eight accident classes. Person-rem per year is the product of the frequency (per year) and the person-rem.

[19] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

Table 10
ANO-1 PRA Release Category Grouping to EPRI Classes

Class	EPRI Description	Frequency	Person-Rem	Person-Rem/yr
1	Intact containment	2.65E-6	1.15E+2	3.05E-4
2	Large containment isolation failures	2.94E-9	4.12E+5	1.21E-3
3a	Small isolation failures (liner breach)	Required		0.00E+0
3b	Large isolation failures (liner breach)	Required		0.00E+0
4	Small isolation failures - failure to seal (type B)	-		
5	Small isolation failures - failure to seal (type C)	-		
6	Containment isolation failures (dependent failure, personnel errors)	4.97E-11	4.20E+4	2.09E-6
7	Severe accident phenomena induced failure (early)	2.12E-7	2.52E+5	5.34E-2
8	Containment bypass	3.97E-8	8.72E+5	3.46E-2
	Total	2.90E-6		8.96E-2

Step 2: Calculate the Type A leakage estimate to define the analysis baseline (3 year test interval)

As displayed in Table 10 the ANO-1 PRA did not identify any release categories specifically associated with EPRI Classes 4 or 5 and the estimate for Class 3 was redistributed back into INTACT. Therefore each of these classes must be evaluated for applicability to this study.

<u>Class 3:</u>

Containment failures in this class are due to leaks such as liner breaches that could only be detected by performing a Type A ILRT. In order to determine the impact of the extended testing interval, the probability of Type A leakage must be calculated.

In order to better assess the range of possible leakage rates, the Class 3 calculation is divided into two classes. Class 3a is defined as a small liner breach and Class 3b is defined as a large liner breach. This division is consistent with the EPRI methology[20]. The calculation of Class 3a and Class 3b probabilities is presented below.

Calculation of Class 3a Probability

Data presented in the EPRI report contains 2 Type A leakage events out of 217 test. Using the data a mean estimate for the probability is determined for Class 3a as shown in Equation 2.

$$p_{Class3a} = \frac{2}{217} = 0.0092 \qquad \text{(eq. 2)}$$

This probability, however, is based on three tests over a 10-year period and not the one per ten-year frequency currently employed at ANO-1[21]. The probability (0.0092) must be adjusted to reflect this difference and is adjusted in step 3 of this calculation.

Multiplying the CDF times the probability of a Class 3a leak develops the Class 3a frequency contribution in accordance with guidance provided by EPRI. The total CDF includes contributions already binned to LERF. To include these contributions would result in a potentially conservative result. Therefore, the LERF contribution from CDF is removed (4.27E-8/yr). The CDF for ANO-1 is 2.90E-6/yr as presented in Table 10 and is adjusted to remove the LERF contribution.

Therefore the frequency of a Class 3a failure is calculated as:

$$FREQ_{class3a} = PROB_{class3a} \times (CDF - Class\ 8 - Class\ 2)$$

$$= 0.0092 \times (2.90E\text{-}6/yr - 3.97E\text{-}8/yr - 2.94E\text{-}9/yr) = 2.64E\text{-}8/yr \qquad \text{(eq. 3)}$$

Calculation of Class 3b Probability

To estimate the failure probability given that no failures have occurred, the guidance provided in the EPRI report[22] suggests the use of a non-informative prior. This approach essentially updates a uniform distribution (no bias) with the available evidence (data) to provide a better estimation of an event.

A beta distribution is typically used for the uniform prior with the parameters α=0.5 and β=1. This is then combined with the existing data (no Class 3b events, 217 tests) using Equation 4.

$$p_{Class3b} = \frac{n+\alpha}{N+\beta} = \frac{0+0.5}{217+1} = \frac{0.5}{217} = 0.0023 \qquad \text{(eq. 4)}$$

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[20] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008

[21] Unit One Integrated Leak Rate Test, Entergy Operations Incorporated, 5120.400, December 17, 2005

[22] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

where: N is the number of tests, n is the number of events (faults) of interest, α, β are the parameters of the non-informative prior distribution. From this solution, the frequency for Class 3b is generated using Equation 5 and is adjusted appropriately to address LERF sequences.

$$FREQ_{class3b} = PROB_{class3b} \times (CDF - Class\ 8 - Class\ 2)$$

$$= 0.0023 \times (2.90E\text{-}6/yr - 3.97E\text{-}8/yr - 2.94e\text{-}9/yr) = 6.56E\text{-}9/yr \qquad \text{(eq. 5)}$$

<u>Class 1:</u>

Although the frequency of this class is not directly impacted by Type A testing and the frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3a and Class 3b in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$$FREQ_{class1} = FREQ_{class1} - (FREQ_{class3a} + FREQ_{class3b}) \qquad \text{(eq. 6)}$$

$$FREQ_{class1} = 2.90E\text{-}6/yr - (2.64E\text{-}8/yr + 6.56E\text{-}9/yr) = 2.62E\text{-}6/yr$$

<u>Class 2:</u>

Class 2 represents large containment isolation failures. Class 2 contains contribution to LERF related to isolation failures without scrubbing credited. The frequency of Class 2 is the sum of those release categories identified in Table 3 as Class 2.

$$FREQ_{class2} = 2.94E\text{-}9/yr \qquad \text{(eq. 7)}$$

<u>Class 4:</u>

This group consists of all core damage accidents for which a failure-to-seal containment isolation failure of Type B test components occurs. By definition, these failures are dependent on Type B testing, and Type A testing will not impact the probability. Therefore this group is not evaluated any further, consistent with the approved methodology.

<u>Class 5:</u>

This group consists of all core damage accidents for which a failure-to-seal containment isolation failure of Type C test components occurs. By definition, these failures are dependent on Type C testing, and Type A testing will not impact the probability. Therefore this group is not evaluated any further, consistent with the approved methodology.

<u>Class 6:</u>

The Class 6 group is comprised of isolation faults that occur as a result of the accident sequence progression. For ANO-1, this class is defined by the ANO-1 SERF category.

$$FREQ_{class6} = 4.97E\text{-}11/yr \qquad \text{(eq. 8)}$$

Class 7:

Class 7 represents early and late containment failure sequences involving phenomena related containment breach. Class 7 contains contributions to LERF related to early release phenomena. The frequency of Class 7 is the sum of those release categories identified in Table 3 as Class 7 and the frequency associated with LATE failures.

$$FREQ_{class7} = 2.12E\text{-}7/yr \qquad \text{(eq. 9)}$$

Class 8:

The frequency of Class 8 is the sum of those release categories identified in Table 3 as Class 8.

$$FREQ_{class8} = 3.97E\text{-}8/yr \qquad \text{(eq. 10)}$$

Table 11 summarizes the above information by the EPRI defined classes. This table also presents dose exposures previously calculated. Class 3a and 3b person-rem values are developed based on the design basis assessment of the intact containment as defined in EPRI's guidance report[23].

The Class 3a and 3b doses are represented as $10L_a$ and $100L_a$ respectively. Table 11 also presents the person-rem frequency data determined by multiplying the failure class frequency by the corresponding exposure.

[23] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

Table 11
Baseline Risk Profile

Class	Description	Frequency (/yr)	Person-rem (calculated)	Person-rem (from L_a factors)	Person-rem (/yr)
1	No containment failure	2.62E-6		1.15E+2	3.01E-4
2	Large containment isolation failures	2.94E-9	4.12E+5		1.21E-3
3a	Small isolation failures (liner breach)	2.64E-8		1.15E+3[2]	3.03E-5
3b	Large isolation failures (liner breach)	6.56E-9		1.15E+4[3]	7.55E-5
4	Small isolation failures - failure to seal (type B)	ε[1]			
5	Small isolation failures - failure to seal (type C)	ε[1]			
6	Containment isolation failures (dependent failure, personnel errors)	4.97E-11	4.20E+4		2.09E-6
7	Severe accident phenomena induced failure (early and late)	2.12E-7	2.52E+5		5.34E-2
8	Containment bypass	3.97E-8	8.72E+5		3.46E-2
	Total	2.90E-6			8.97E-2

1. ε represents a probabilistically insignificant value.
2. 10 times L_a.
3. 100 times L_a.

The percent risk contribution due to Type A testing is defined as follows:

$$\%Risk_{BASE} = [(Class3a_{BASE} + Class3b_{BASE}) / Total_{BASE}] \times 100 \qquad \text{(eq. 11)}$$

Where:

$Class3a_{BASE}$ = Class 3a person-rem/year = 3.03E-5 person-rem/year

$Class3b_{BASE}$ = Class 3b person-rem/year = 7.55E-5 person-rem/year

$Total_{BASE}$ = total person-rem year for baseline interval = 8.97E-2 person-rem/year (Table 11)

$\%Risk_{BASE}$ = [(3.03E-5 + 7.55E-5) / 8.96E-2] x 100 = **0.118%** (eq. 12)

Step 3: Calculate the Type A leakage estimate to address the current inspection interval

The current surveillance testing requirement for Type A testing and allowed by 10 CFR 50, Appendix J is at least once per 10 years based on an acceptable performance history (defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than $1.0L_a$).

According to the ERRI report[24], extending the Type A ILRT interval from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months. Multiplying the testing interval by 0.5 and multiplying by 12 to convert from "years" to "months" calculates the average time for an undetected condition to exist.

The increase for a 10-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from 18 months to 60 months) multiplied by the existing Class 3a probability as shown in Equation 13.

$$p_{Class3a}(10y) = 0.0275 \times \left(\frac{60}{18}\right) = 0.0916 \qquad \text{(eq. 13)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 14.

$$p_{Class3b}(10y) = 0.00273 \times \left(\frac{60}{18}\right) = 0.0091 \qquad \text{(eq. 14)}$$

Risk Impact due to 10-year Test Interval

Based on the approved EPRI methodology and the NEI guidance[25], the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences.

The risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage. Additionally the Class 1 frequency is adjusted to maintain the overall core damage frequency constant. The results of this calculation are presented in Table 12 below.

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[24] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

[25] Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 0, Nuclear Energy Institute, NEI 94-01, July 26, 1995.

Table 12
Risk Profile for Once in Ten Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure[1]	2.54E-6	1.15E+2	2.92E-4
2	Large Containment Isolation Failures	2.94E-9	4.12E+5	1.21E-3
3a	Small Isolation Failures (Liner breach)	8.79E-8	1.15E+3	1.01E-4
3b	Large Isolation Failures (Liner breach)	2.19E-8	1.15E+4	2.52E-4
4	Small isolation failures - failure to seal (type B)	ε[3]		
5	Small isolation failures - failure to seal (type C)	ε[3]		
6	Containment Isolation Failures (dependent failure, personnel errors)	4.97E-11	4.20E+4	2.09E-6
7	Severe Accident Phenomena Induce Failure (Early and Late)	2.12E-7	2.52E+5	5.34E-2
8	Containment Bypass	3.97E-8	8.72E+5	3.46E-2
	Total	2.90E-6		8.99E-2

1. The PRA frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 7.
3. ε represents a probabilistically insignificant value.

Using the same methods as for the baseline, and the data in Table 8 the percent risk contribution due to Type A testing is as follows:

$$\%Risk_{10} = [(Class3a_{10} + Class3b_{10}) / Total_{10}] \times 100 \qquad \text{(eq. 15)}$$

Where:

$Class3a_{10}$ = Class 3a person-rem/year = 1.01E-4 person-rem/year

$Class3b_{10}$ = Class 3b person-rem/year = 2.52E-4 person-rem/year

$Total_{10}$ = total person-rem year for current 10-year interval = 8.99E-2 person-rem/year (Table 12)

$$\%Risk_{10} = [(1.01E\text{-}4 + 2.52E\text{-}4) / 8.99E\text{-}2] \times 100 = \mathbf{0.392\%} \qquad \text{(eq. 16)}$$

The percent risk increase ($\Delta\%Risk_{10}$) due to a ten-year ILRT over the baseline case is as follows:

$$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100.0 \qquad \text{(eq. 17)}$$

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 8.97E-2 person-rem/year (Table 11)

$Total_{10}$ = total person-rem/year for 10-year interval = 8.99E-2 person-rem/year (Table 12)

$$\Delta\%Risk_{10} = [(8.99E\text{-}2 - 8.97E\text{-}2) / 8.97E\text{-}2] \times 100.0 = \mathbf{0.266\%} \qquad \text{(eq. 18)}$$

Step 4: Calculate the Type A leakage estimate to address extended inspection intervals

If the test interval is extended to 1 per 15 years, the average time that a leak detectable only by an ILRT test goes undetected increases to 90 months (0.5 x 15 x 12). For a 15-yr-test interval, the result is the ratio (90/18) of the exposure times as was the case for the 10 year case. Increasing the ILRT test interval from once every 3 years to once per 15 years results in a proportional increase in the overall probability of leakage.

The approach for developing the risk contribution for a 15-year interval is the same as that for the 10-year interval. The increase for a 15-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from 18 months to 90 months) multiplied by the existing Class 3a probability as shown in Equation 19.

$$p_{Class3a}(15y) = 0.0275 \times \left(\frac{90}{18}\right) = 0.1374 \qquad \text{(eq. 19)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 20.

$$p_{Class3b}(15y) = 0.00273 \times \left(\frac{90}{18}\right) = 0.0137 \qquad \text{(eq. 20)}$$

Risk Impact due to 15-year Test Interval

As stated for the 10-year case, the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences.

The increased risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage. Additionally the Class 1 frequency is adjusted to maintain the overall core damage frequency constant. The results of this calculation are presented in Table 13 below.

Table 13
Risk Profile for Once in Fifteen Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure[1]	2.49E-6	1.15E+2	2.86E-4
2	Large Containment Isolation Failures	2.94E-9	4.12E+5	1.21E-3
3a	Small Isolation Failures (Liner breach)	1.32E-7	1.15E+3	1.52E-4
3b	Large Isolation Failures (Liner breach)	3.28E-8	1.15E+4	3.77E-4
4	Small isolation failures - failure to seal (type B)	ε[3]		
5	Small isolation failures - failure to seal (type C)	ε[3]		
6	Containment Isolation Failures (dependent failure, personnel errors)	4.97E-11	4.20E+4	2.09E-6
7	Severe Accident Phenomena Induce Failure (Early and Late)	2.12E-7	2.52E+5	5.34E-2
8	Containment Bypass	3.97E-8	8.72E+5	3.46E-2
	Total	2.90E-6		9.01E-2

1. The PRA frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 7.
3. ε represents a probabilistically insignificant value.

Using the same methods as for the baseline, and the data in Table 9 the percent risk contribution due to Type A testing is as follows:

$$\%Risk_{15} = [(Class3a_{15} + Class3b_{15}) / Total_{15}] \times 100 \qquad \text{(eq. 21)}$$

Where:

$Class3a_{15}$ = Class 3a person-rem/year = 1.52E-4 person-rem/year

$Class3b_{15}$ = Class 3b person-rem/year = 3.77E-4 person-rem/year

$Total_{15}$ = total person-rem year for 15-year interval = 9.01E-2 person-rem/year (Table 13)

$$\%Risk_{15} = [(1.52E\text{-}4 + 3.77E\text{-}4) / 9.01E\text{-}2] \times 100 = \mathbf{0.587\%} \qquad \text{(eq. 22)}$$

The percent risk increase ($\Delta\%Risk_{15}$) due to a fifteen-year ILRT over the baseline case is as follows:

$$\Delta\%Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100.0 \qquad \text{(eq. 23)}$$

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 8.97E-2 person-rem/year (Table 11)

$Total_{15}$ = total person-rem/year for 15-year interval = 9.01E-2 person-rem/year (Table 13)

$$\Delta\%Risk_{15} = [(9.01E\text{-}2 - 8.97E\text{-}2) / 8.97E\text{-}2] \times 100.0 = \mathbf{0.455\%} \qquad \text{(eq. 24)}$$

Step 5: Calculate increase in risk due to extending Type A inspection intervals

Based on the guidance in the EPRI guidance document[26], the percent increase in the total integrated plant risk from a fifteen-year ILRT over a current ten-year ILRT is computed as follows:

$$\%Total_{10\text{-}15} = [(Total_{15} - Total_{10}) / Total_{10}] \times 100 \qquad \text{(eq. 25)}$$

Where:

$Total_{10}$ = total person-rem/year for 10-year interval = 8.99E-2 person-rem/year (Table 12)

$Total_{15}$ = total person-rem/year for 15-year interval = 9.01E-2 person-rem/year (Table 13)

$$\%\ Total_{10\text{-}15} = [(9.01E\text{-}2 - 8.99E\text{-}2) / 8.99E\text{-}2] \times 100 = \mathbf{0.189\%} \qquad \text{(eq. 26)}$$

Step 6: Calculate the change in Risk in terms of Large Early Release Frequency (LERF)

The risk impact associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from containment could in fact result in a larger release due to failure to detect a pre-existing leak during the relaxation period.

From the EPRI Report, the Class 3a dose is assumed to be 10 times the allowable intact containment leakage, L_a (or 1,150 person-rem) and the Class 3b dose is assumed to be 100 times L_a (or 11,500 person-rem). The method for defining the dose equivalent for allowable leakage (L_a) is developed in the EPRI report. This compares to a historical observed average of twice L_a. Therefore, the estimate is somewhat conservative.

Based on EPRI guidance, only Class 3 sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because for these sequences the containment remains intact. Therefore, the

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[26] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

containment leak rate is expected to be small (less than $2L_a$). A larger leak rate would imply an impaired containment, such as Classes 2, 3, 6 and 7. Late releases are excluded regardless of the size of the leak because late releases are, by definition, not a LERF event.

Therefore, the change in the frequency of Class 3b sequences is used as the increase in LERF for ANO-1, and the change in LERF can be determined by the differences. The EPRI guidance document[27] identifies that Class 3b is considered to be the contributor to LERF. Table 10 summarizes the results of the LERF evaluation that Class 3b is indicative of a LERF sequence.

Table 14
Impact on LERF due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	6.56E-9/yr	2.19E-8/yr	3.28E-8/yr
∆LERF (3 year baseline)		1.53E-8/yr	2.63E-8/yr
∆LERF (10 year baseline)			1.09E-8/yr

Reg. Guide 1.174[28] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. The EPRI report cites Reg. Guide 1.174 and defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr. Since the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

By increasing the ILRT interval from the currently acceptable 10 years to a period of 15 years results in an increase in the contribution to LERF of 1.09E-8/yr. This value meets the guidance in Reg. Guide 1.174 defining very small changes in LERF. The LERF increase measured from the original 3-in-10-year interval to the 15-year interval is 2.63E-8/yr, which is less than the criterion presented in Regulatory Guide 1.174.

Step 7: Calculate the change in Conditional Containment Failure Probability (CCFP)

The conditional containment failure probability (CCFP) is defined as the probability of containment failure given the occurrence of an accident. This probability can be expressed using the following equation:

$$CCFP = 1 - \left[\frac{f(ncf)}{CDF} \right] \quad \text{(eq. 27)}$$

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27 Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

28 An Approach for Using Probabilistic Risk Assessment in Risk-Informed decisions on Plant-Specific Changes to the Licensing Basis, U.S. Nuclear Regulatory Commission (USNRC), Regulatory Guide 1.174, July 1998.

Where *f(ncf)* is the frequency of those sequences which result in no containment failure. This frequency is determined by summing the Class 1 and Class 3a results, and CDF is the total frequency of all core damage sequences.

Therefore the change in CCFP for this analysis is the CCFP using the results for 15 years ($CCFP_{15}$) minus the CCFP using the results for 10 years ($CCFP_{10}$). This can be expressed by the following:

$$\Delta CCFP_{10-15} = CCFP_{15} - CCFP_{10} \quad \text{(eq. 28)}$$

Using the data previously developed the change in CCFP from the current testing interval is calculated and presented in Table 15.

Table 15
Impact on Conditional Containment Failure Probability due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	2.64E-6	2.62E-6	2.61E-6
f(ncf)/CDF	0.910	0.903	0.899
CCFP	8.99E-2	9.52E-2	9.90E-2
ΔCCFP (3 year baseline)		5.27E-3	9.04E-3
ΔCCFP (10 year baseline)			3.77E-3

5.0 SENSITIVITY STUDIES

This appendix provides sensitivity studies suggested in the EPRI report[29] for the ANO-1 ILRT extension assessment. This includes an evaluation of assumptions made in relation to liner corrosion, the use of the expert elicitation, and the impact of external events.

5.1 LINER CORROSION

The analysis approach utilizes the Calvert Cliffs Nuclear Plant (CCNP) methodology[30] as modified by EPRI. This methodology is an acceptable approach to incorporate the liner corrosion issue into the integrated leak rate test (ILRT) extension risk evaluation. The results of the analysis indicate that increasing the interval from three years to fifteen years did not significantly increase plant risk of a large early release.

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29 Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

30 Letter to NRC from Calvert Clifffs Nuclear Power Plant Unit No.1. Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, dated March 27, 2002.

Table 16 summarizes the results obtained from the CCNP methodology utilizing plant-specific data for ANO-1.

Table 16
ANO-1 Liner Corrosion Risk Assessment Results Using CCNP Methodology

<table>
<tr><th>Step</th><th>Description</th><th colspan="2">Containment Cylinder and Dome (85%)</th><th colspan="2">Containment Basemat (15%)</th></tr>
<tr><td>1</td><td>Historical liner flaw likelihood
Failure data: containment location specific
Success data: based on 70 steel-lined containments and 5.5 years since the 10CFR 50.55a requirements of periodic visual inspections of containment surfaces</td><td colspan="2">Events 2
(Brunswick 2 and North Anna 2)
2 / (70 x 5.5) = 5.19E-03</td><td colspan="2">Events: 0
Assume a half failure
0.5 / (70 x 5.5) = 1.30E-03</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Aged adjusted liner flaw likelihood
During the 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the 5th to 10th year set to the historical failure rate.</td><td>Year
1
average 5-10
15</td><td>Failure rate
2.05E-03
5.19E-03
1.43E-02</td><td>Year
1
average 5-10
15</td><td>Failure rate
5.13E-04
1.30E-03
3.57E-03</td></tr>
<tr><td colspan="2">15 year average = 6.44E-03</td><td colspan="2">15 year average = 1.61E-03</td></tr>
<tr><td>3</td><td>Increase in flaw likelihood between 3 and 15 years
Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years.</td><td colspan="2">0.73% (1 to 3 years)
4.18% (1 to 10 years)
9.66% (1 to 15 years)</td><td colspan="2">0.18% (1 to 3 years)
1.04% (1 to 10 years)
2.41% (1 to 15 years)</td></tr>
<tr><td>4</td><td>Likelihood of breach in containment given liner flaw</td><td colspan="2">1%</td><td colspan="2">0.1%</td></tr>
<tr><td>5</td><td>Visual inspection detection failure likelihood</td><td colspan="2">10%
5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT)
All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.</td><td colspan="2">100%
Cannot be visually inspected</td></tr>
<tr><td>6</td><td>Likelihood of non-detected containment leakage
(Steps 3 x 4 x 5)</td><td colspan="2">0.00073% (3 years)
0.73% x 1% x 10%
0.00418% (10 years)
4.18% x 1% x 10%
0.00966% (15 years)
9.66% x 1% x 10%</td><td colspan="2">0.000180% (3 years)
0.18% x 0.1% x 100%
0.00104% (10 years)
1.04% x 0.1% x 100%
0.00241% (15 years)
2.41% x 0.1% x 100%</td></tr>
</table>

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for containment cylinder and dome and the containment basemat.

Total likelihood of non-detected containment leakage (3 yr) = 0.00073% + 0.000180% = 0.00091%

Total likelihood of non-detected containment leakage (10 yr) = 0.00418% + 0.00104% = 0.00522%

Total likelihood of non-detected containment leakage (15 yr) = 0.00966% + 0.00241% = 0.01207%

This likelihood is then multiplied by the non-LERF containment failures for ANO-1. This value is calculated by the following equation for each period of interest. LERF is comprised of Class 8 and Class 3b cases (Equation 29).

$$\text{Non-LERF} = \text{CDF} - \text{Class 2} - \text{Class 8} - \text{Class 3b} \qquad \text{(eq. 29)}$$

A final adjustment can be made to address cases with containment spray operation. It is conservatively not addressed and would not substantially alter the overall results. Table 17 presents the data and the resultant increase in LERF due to liner corrosion for each case.

Table 17
Liner Corrosion LERF Adjustment Using CCNP Methodology

Case	CDF (/yr)	Class 2 (/yr)	Class 8 (/yr)	Class 3b (/yr)	Likelihood of Non-detected Corrosion Leakage	Increase in LERF (/yr)
3-years	2.90E-6	2.94E-9	3.97E-8	6.56E-9	9.10E-6	2.60E-11
10-years	2.90E-6	2.94E-9	3.97E-8	2.19E-8	5.22E-5	1.48E-10
15-years	2.90E-6	2.94E-9	3.97E-8	3.28E-8	1.21E-4	3.42E-10

This contribution is added to the Class 3b LERF cases and the sensitivity analysis performed. Table 18 provides a summary of the base case as well as the corrosion sensitivity case. The “Delta Person-Rem” column provides the change in person-rem between the case without corrosion and the case that considers corrosion. Values within parentheses “( )” indicate the change or delta between the without corrosion and corrosion cases.

Table 18
ANO-1 Summary of Base Case and Corrosion Sensitivity Cases

EPRI Class	Base Case (3 per 10 years)					1 per 10 years					1 per 15 years				
	Without Corrosion		With Corrosion			Without Corrosion		With Corrosion			Without Corrosion		With Corrosion		
	Frequency	Person-rem per year	Frequency	Person-rem per year	Delta Person-Rem per year	Frequency	Person-rem per year	Frequency	Person-rem per year	Delta Person-Rem per year	Frequency	Person-rem per year	Frequency	Person-rem per year	Delta Person-Rem per year
1	2.62E-6	3.01E-4	2.62E-6	3.01E-4	-2.99E-9	2.54E-6	2.92E-4	2.54E-6	2.92E-4	-1.70E-8	2.49E-6	2.86E-4	2.48E-6	2.86E-4	-3.94E-8
2	2.94E-9	1.21E-3	2.94E-9	1.21E-3	n/a	2.94E-9	1.21E-3	2.94E-9	1.21E-3	n/a	2.94E-9	1.21E-3	2.94E-9	1.21E-3	n/a
3a	2.64E-8	3.03E-5	2.64E-8	3.03E-5	n/a	8.79E-8	1.01E-4	8.79E-8	1.01E-4	n/a	1.32E-7	1.52E-4	1.32E-7	1.52E-4	n/a
3b	6.56E-9	7.55E-5	6.59E-9	7.58E-5	2.99E-7	2.19E-8	2.52E-4	2.20E-8	2.53E-4	1.70E-6	3.28E-8	3.77E-4	3.32E-8	3.81E-4	3.94E-6
6	4.97E-11	2.09E-6	4.97E-11	2.09E-6	n/a	4.97E-11	2.09E-6	4.97E-11	2.09E-6	n/a	4.97E-11	2.09E-6	4.97E-11	2.09E-6	n/a
7	2.12E-7	5.34E-2	2.12E-7	5.34E-2	n/a	2.12E-7	5.34E-2	2.12E-7	5.34E-2	n/a	2.12E-7	5.34E-2	2.12E-7	5.34E-2	n/a
8	3.97E-8	3.46E-2	3.97E-8	3.46E-2	n/a	3.97E-8	3.46E-2	3.97E-8	3.46E-2	n/a	3.97E-8	3.46E-2	3.97E-8	3.46E-2	n/a
CDF	2.90E-6	8.97E-2	2.90E-6	8.97E-2	2.96E-7	2.90E-6	8.99E-2	2.90E-6	8.99E-2	1.69E-6	2.90E-6	9.01E-2	2.90E-6	9.01E-2	3.90E-6
Class 3b LERF	6.56E-9		6.59E-9 (2.60E-11)			2.19E-8		2.20E-8 (1.48E-10)			3.28E-8		3.32E-8 (3.42E-10)		
Delta LERF (from base case of 3 per 10 years)						1.53E-8		1.54E-8 (1.22E-10)			2.63E-8		2.66E-8 (3.16E-10)		
Delta LERF from 1 per 10 years						N/A					1.09E-8		1.11E-8 (1.94E-10)		

The inclusion of corrosion does not result in an increase in LERF sufficient to invalidate the baseline analysis and the overall impact is negligible.

5.2 DEFECT SENSITIVITY AND EXPERT ELICIATION SENSITIVITY

A second sensitivity case on the impacts of assumptions regarding pre-existing containment defect or flaw probabilities of occurrence and magnitude, or size of the flaw, is performed as described in the EPRI guidance document[1]. In this sensitivity case, an expert elicitation was conducted to develop probabilities for pre-existing containment defects that would be detected by the ILRT only based on the historical testing data.

Using the expert knowledge, this information was extrapolated into a probability versus magnitude relationship for pre-existing containment defects. The failure mechanism analysis also used the historical ILRT data augmented with expert judgment to develop the results. Details of the expert elicitation process and results are contained in the EPRI report. The expert elicitation process has the advantage of considering the available data for small leakage events, which have occurred in the data, and extrapolate those events and probabilities of occurrence to the potential for large magnitude leakage events.

The expert elicitation results are used to develop sensitivity cases for the risk impact assessment. Employing the results requires the application of the ILRT interval methodology using the expert elicitation to change in the probability of pre-existing leakage in the containment.

The baseline assessment uses the Jefferys non-informative prior and the expert elicitation sensitivity study uses the results of the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency, can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the basic methodology (i.e., 10 La for small and 100 La for large) are used here. Table 19 presents the magnitudes and probabilities associated with the Jefferys non-informative prior and the expert elicitation use in the base methodology and this sensitivity case.

Table 19
ANO-1 Summary of ILRT Extension Using Expert Elicitation Values

Leakage Size (L_a)	Jefferys Non-Informative Prior	Expert Elicitation Mean Probability of Occurrence	Percent Reduction
10	2.70E-02	3.88E-03	86%
100	2.70E-03	9.86E-04	64%

[1] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

Taking the baseline analysis and using the values provided in Table 19 for the expert elicitation yields the results in Table 20 are developed.

Table 20
ANO-1 Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
1	2.65E-6	2.64E-6	1.15E+2	3.03E-4	2.61E-6	3.00E-4	2.59E-6	2.98E-4
2	2.94E-9	2.94E-9	4.12E+5	1.21E-3	2.94E-9	1.21E-3	2.94E-9	1.21E-3
3a	N/A	1.11E-8	1.15E+3	1.28E-5	3.71E-8	4.26E-5	5.56E-8	6.39E-5
3b	N/A	7.08E-10	1.15E+4	8.14E-6	2.36E-9	2.71E-5	3.54E-9	4.07E-5
6	4.97E-11	4.97E-11	4.20E+4	2.09E-6	4.97E-11	2.09E-6	4.97E-11	2.09E-6
7	2.12E-7	2.12E-7	2.52E+5	5.34E-2	2.12E-7	5.34E-2	2.12E-7	5.34E-2
8	3.97E-8	3.97E-8	8.72E+5	3.46E-2	3.97E-8	3.46E-2	3.97E-8	3.46E-2
Totals	2.90E-6	2.90E-6	1.59E+6	8.96E-2	2.90E-6	8.96E-2	2.90E-6	8.97E-2
Δ LERF (3 per 10 yrs base)	N/A				1.65E-9		2.83E-8	
Δ LERF (1 per 10 yrs base)					N/A		1.18E-9	
CCFP	8.79%				8.85%		8.89%	

The results illustrate how the expert elicitation reduces the overall change in LERF and the overall results are more favorable with regard to the change in risk.

5.3 POTENTIAL IMPACTS FROM EXTERNAL EVENTS

An assessment of the impact of external events is performed. The primary basis for this investigation is the determination of the total LERF following an increase in the ILRT testing interval from 3 in 10 years to 1 in 15 years.

External events were evaluated in the ANO-1 Individual Plant Examination of External Events (IPEEE)[2]. The IPEEE program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and an understanding of severe accident risk. The primary areas of external event analysis for the ANO-1 IPEEE were seismic and internal fires, and other external events. Seismic and fire were considered to be the most limiting due to their frequency of occurrence and their potential impact on plant operability. Therefore it is assumed that they bound the risk contribution from other external events. Both seismic and internal fire were examined but the analysis contained conservative assumptions related to consequential failures due to external events such that the absolute CDF is considered an understatement of plant performance and an over estimation of CDF.

Seismic events were addressed through a Seismic Margin Analysis (SMA) as part of the IPEEE for ANO-1. The Seismic PRA method screened all the components that met a high confidence low probability of failure (HCLPF) for the review level seismic event occurring with a magnitude of 0.3g. The remaining components were grouped together as a proxy component. It was assumed that if this proxy component failed it would result in core damage. This method is considered conservative.

The SMA information is used in conjunction with the improvements that have been incorporated into the internal event model since the IPEEE was performed. Prior seismic analyses have indicated that for a well-designed plant, seismic contributions are a combination of low acceleration events with random failures and higher acceleration events with dependent component or structural failures due to forces associated with the seismic event.

As cited in NUREG-1742[3], the controlling failure typically involves prolonged loss of ac power leading to a station blackout. Low acceleration events lead to a disruption of offsite power sources and result in a prolonged need for onsite sources. This contribution has been estimated utilizing the current internal events analysis and based on the loss of offsite power (LOSP) initiating events analysis to define a conditional core damage probability (CCDP). This value is then combined with a typical estimation for the median capacity of the offsite power supply (0.3g, median capacity). The frequency is multiplied by 0.5 for the likelihood of failure of offsite sources given a seismic event.

The CCDP is calculated by modifying the ANO-1 CAFTA model[4] to only calculate the CCDP associated with loss of offsite power scenarios. The model contains three unique IEs that are associated with LOSP and are described as grid related, plant center, and weather related. Since the impact of any of the three initiating events is the same, only one event (%T3) is set to a value of 1.0 to represent a condition reflecting a loss of offsite power and the quantification yields the CCDP due to LOSP. The quantification assumes that offsite power cannot be restored within 24 hours. Since the standard recovery techniques utilize non-seismic data, it is not applicable. The calculated CCDP for SBO without recovery is 2.26E-3. From the seismic

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[2] Stevenson & Associates, IPEEE Seismic Margins Assessment (SMA) of Arkansas Nuclear One, Unit 1, Entergy, 96-R-1006-02, April 1996.

[3] Perspectives Gained from the IPEEE Program, USNRC, NUREG-1742, April 2002. Other Reference Documentation

[4] ANO-1 CAFTA model modified for the ILRT analysis (additional file details provided in the Reference section)

hazard curve[5], a 0.3g seismic event has a median frequency of 9.09E-5/yr. At this seismicity level, the best estimate fragility for loss of power yields a probability of 0.5. Combining the frequency, the CCDP and the probability of losing offsite power yields an estimate for the frequency contribution for low acceleration seismic events. The seismic frequency estimate is 1.03E-7/yr.

In addition to the prolonged loss of offsite power case, at higher accelerations the seismic forces result in component and/or structural concerns. For most safety-related components, the structures are not limiting and the impact can be based on component-level fragility. Reference 17 utilized existing seismic fragility information to arrive at a generic estimate for component capacities. A review of this report indicates that major equipment exhibits at least 1.0g median capacity given standard assumptions related to anchorage and location.

To develop an estimate for the seismic failures for ANO-1 we utilize a median capacity of 1.0g. The corresponding recurrence frequency of seismic of this acceleration or greater is 4.8E-6/yr. This is again multiplied by a mean fragility of 0.5 to arrive at a value of 2.4E-6/yr. This represents the frequency of core damage due to seismically-induced component and/or structural failure.

This estimate is considered a bounding contribution for seismically induced failures, because the probability of a seismically induced component failure associated with a seismic event of this magnitude would dominate postulated random failure probability. A typical assumption of one-fails-all-fail typically assumed for seismic faults would also tend to defeat redundant components and again lead to the conclusion that for this seismicity range the seismic failures would provide a reasonable estimate for the contribution to core damage and LERF.

Summing the estimates for lower acceleration seismic events which would be dominated by prolonged station blackout with the contribution from higher acceleration seismic events involving seismically induced component failures yields an estimated CDF contribution of 2.50E-6/yr (2.4E-6/yr + 1.03E-7/yr) and is controlled by higher acceleration seismic initiating event.

The findings contained in NUREG-1742[6] indicate that the fire CDF is primarily determined by plant transient type of events such as those from assessed plant transients. The judgment is made based on this observation that it is reasonable to assume that the ratio of intact to impaired containments will be similar for fire as for the internal events such that the total CDF and the breakdown by EPRI Class will be equivalent to that presented for the internal events.

For ANO-1 internal events the total adjusted CDF is 2.90E-6/yr. The associated LERF contribution to Class 8 is 5.82E-8/yr and is comprised of SGTR and ISLOCA. Since SGTR and ISLOCA are unique initiators removing them from the CDF provides an approximate value for a refined internal fire assessment. Conservatively Large LOCA and Medium LOCA are retained.

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5 "Arkansas Hazard.xls", Excel file provided by Richard Harris of Entergy to Stephen Pionke of RSC via e-mail on Jan 8, 2010, 11:47 AM.

6 Perspectives Gained from the IPEEE Program, USNRC, NUREG-1742, April 2002. Other Reference Documentation

Per the guidance contained in the EPRI report[7] the figure-of-merit for the risk impact assessment of extended ILRT intervals is given as:

delta LERF = The change in frequency of Accident Class 3b

Using the percentage of total CDF contributing to LERF for the fire, seismic, and other external events as an approximation for the early CDF applicable to EPRI Accident Class 3b yields the following:

CDF_{FIRE} = 2.90E-6/yr – 5.82E-8/yr = 2.84E-6/yr (eq. 30)

$CDF_{SEISMIC}$ = 2.4E-6/yr + 1.03E-7/yr = 2.5E-6/yr (eq. 31)

Class 3b Frequency = [(CDF_{FIRE}) + ($CDF_{SEISMIC}$)] * Class 3b Leakage Probability

Class 3b Frequency = [(2.84E-6/yr) + (2.5E-6/yr)] * 2.3E-03 = 1.22E-8/yr

No adjustment is made to the CDF values since LERF sequences are typically associated with SGTR or interfacing system LOCA sequences which are not represented by the external event assessments. This is potentially conservative, but is reasonable based on the simplified assessment, the conservative nature of the external events studies and the fact that many of the external event scenarios are long term station blackout and long term level of analysis detail. The change in LERF is estimated for the 1 in 10 year and 1 in 15 year cases and the change defined for the external events in Table 21.

Table 21
ANO-1 Upper Bound External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 1 per 10 years)
	3 per 10 year	1 per 10 year	1 per 15 year	
External Events	1.22E-8	4.08E-8	6.12E-8	4.90E-8
Internal Events	6.57E-9	2.19E-8	3.28E-8	2.63E-8
Combined	1.88E-8	6.27E-8	9.41E-8	7.53E-8

The internal event results are also provided to allow a composite value to be defined. When both the internal and external event contributions are combined the total change in LERF does not exceed the guidance for very small change in risk and does not exceed the 1.0E-7/yr change in LERF. The LERF increase supports the conclusion that the increased duration between tests does not result in a significant change in risk and the increase is acceptable per the criterion defined in the EPRI guidance document[8].

[7] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

[8] Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

6.0 REFERENCES

1. An Approach for Using Probabilistic Risk Assessment in Risk-Informed decisions on Plant-Specific Changes to the Licensing Basis, U.S. Nuclear Regulatory Commission (USNRC), Regulatory Guide 1.174, July 1998.

2. ANO-1 Large Early Release Frequency (LERF) Model, Rev. 1, Entergy Operations Incorporated, PRA-A2-01-001S12, June 2009.

3. Appendix J to Part 50 – Primary Reactor Containment Leakage Testing for Water-Cooling Power Reactors, U.S. Nuclear Regulatory Commission (USNRC), 10 CFR Part 50, Appendix J, January 2006.

4. "Arkansas Hazard.xls", Excel file provided by Deepak Rao of Entergy to Robert E Burrell of Entergy via e-mail on June 6, 2013, 10:03 AM.

5. Breeding, R. J., et al, Evaluation of Severe Accident Risks: Surry Unit 1, Main Report, Rev. 1, U.S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 3, October 1990.

6. Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.

7. Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 0, Nuclear Energy Institute, NEI 94-01, July 26, 1995.

8. Letter to NRC from Calvert Clifffs Nuclear Power Plant Unit No.1. Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, dated March 27, 2002.

9. Performance-Based Containment Leak-Test Program, USNRC, NUREG-1493, July 1995.

10. Perspectives Gained from the IPEEE Program, USNRC, NUREG-1742, April 2002. Other Reference Documentation

11. Reactor Risk Reference Document, Appendices J-O, Draft for Comment, United States Nuclear Regulatory Commission (USNRC), NUREG-1150, January 1987.

12. Stevenson & Associates, IPEEE Seismic Margins Assessment (SMA) of Arkansas Nuclear One, Unit 1, Entergy, 96-R-1006-02, April 1996.

13. Unit One Integrated Leak Rate Test, Entergy Operations Incorporated, 5120.400, December 17, 2005.

14. ANO-1 at Power Fault Tree Model

ANO-1 at Power Fault Tree Model	Size	Date
A1R4P0EOS.caf	407 KB	5/28/2013 12:40 pm
A1R4P0EOS.rr	10072 KB	6/3/2013 10:37 am
1FLG04p0.caf	10 KB	5/28/2013 12:40 pm
A1rul4p00(ILRT).txt	40 KB	5/31/2013 10:16 pm
1mut04p0.txt	9 KB	5/28/2013 12:40 pm

15. ANO-1 at Power CUTSET files

ANO-1 at Power CUTSET files	Size	Date
COREDAMAGE.CUT	725 KB	7/11/2013 10:41 am

Attachment 7 to

1CAN121302

List of Regulatory Commitments

List of Regulatory Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are not commitments.

COMMITMENT	**TYPE (Check one)**		**SCHEDULED COMPLETION DATE (If Required)**
	ONE-TIME ACTION	**CONTINUING COMPLIANCE**	
ANO-1 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.		X	Following the NRC approval of this license amendment request.
ANO-1 will provide the information as requested in Condition 1 listed in the SE for NEI 94-01, Revision 3-A if a Type C frequency is extended.		X	Following the NRC approval of this license amendment request.
ANO-1 will provide the information as requested in Condition 2 listed in the SE for NEI 94-01, Revision 3-A if a Type C frequency is extended.		X	Following the NRC approval of this license amendment request.