AmerenUE
Callaway Plant

PO Box 620
Fulton, MO 65251

March 20, 2009

ULNRC-05598

U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

10 CFR 50.90

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
PROPOSED REVISION TO TECHNICAL SPECIFICATION 5.5.16
"CONTAINMENT LEAKAGE RATE TESTING PROGRAM"
(LICENSE AMENDMENT REQUEST LDCN 09-0008)

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," AmerenUE (Union Electric) hereby requests an amendment to the Facility Operating License (NPF-30) for Callaway Plant in order to incorporate a proposed change to Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," which establishes the program for leakage rate testing of the containment, as required by 10 CFR 50.54, "Conditions of licenses," Section (o) and 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, "Performance Based Requirements," as modified by approved exemptions.

Specifically, AmerenUE proposes to revise TS 5.5.16 to reflect a one-time five-year deferral of the containment Type A integrated leak rate test from once in ten years to once in 15 years. The proposed amendment is risk-informed and follows the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Rick Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Bases." AmerenUE has performed an analysis showing that the increase in risk resulting from the proposed amendment is small and within established guidance. AmerenUE has also determined that defense-in-depth principles will be maintained based on risk and other considerations.

Essential or supporting information for the proposed TS change is provided in the attachments to this letter. Attachment 1 provides a detailed description and technical evaluation of the proposed change, including AmerenUE's determination that the proposed change involves no significant hazards consideration. Attachment 2 provides mark-ups of the current, affected TS pages to show the proposed change, and Attachment 3 provides a copy of the affected TS pages retyped with the proposed change incorporated (if approved).

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AmerenUE has assessed the risk implications of a one-time five-year deferral of the Callaway Plant Type A test interval from once in ten years to once in 15 years. The evaluation indicated that the analyzed Integrated Leak Rate Test (ILRT) interval extension has a minimal impact on public risk. The risk assessment supporting the proposed change is included in Attachment 4. In addition, the extension of the ILRT interval is consistent with extensions recently granted to other licensees.

The Callaway Plant On-site Review Committee and the Nuclear Safety Review Board have reviewed and approved this amendment application. In addition, it has been determined that this amendment application involves no significant hazards consideration as determined per 10 CFR 50.92, "Issuance of amendment," and that pursuant to 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Section (b) no environmental assessment is required to be prepared in connection with issuance of this amendment. It should also be noted that pursuant to 10 CFR 50.91, "Notice for public comment; State consultation," Section (b)(1), AmerenUE is providing the State of Missouri with a copy of this proposed amendment.

AmerenUE respectfully requests approval of the proposed license amendment by March 31, 2010. Once approved, it is anticipated that the requested license amendment will be required to be implemented within 90 days from the issue date of the amendment.

For any questions on the above or attached, please contact Tom Elwood, Supervising Engineer, Regulatory Affairs and Licensing at (573) 676-6479.

I declare under penalty of perjury that the foregoing and attached are true and correct.

Sincerely,

Executed on: March 20, 2009

L. Scott Sandbothe
Manager, Regulatory Affairs

KRA/nls

Attachments:
1) Evaluation
2) Marked-up Technical Specification Pages
3) Retyped Technical Specification Pages
4) Risk Assessment
5) PRA Technical Adequacy Report

cc: U.S. Nuclear Regulatory Commission (Original and 1 copy)
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

Mr. Elmo E. Collins, Jr.
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
612 E . Lamar Blvd., Suite 400
Arlington, TX 76011-4125

Senior Resident Inspector
Callaway Resident Office
U.S. Nuclear Regulatory Commission
8201 NRC Road
Steedman, MO 65077

Mr. Mohan C. Thadani (2 copies)
Licensing Project Manager, Callaway Plant
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-8G14
Washington, DC 20555-2738

Hardcopy:

Certrec Corporation
4200 South Hulen, Suite 422
Fort Worth, TX 76109
(Certrec receives ALL attachments as long as they are non-safeguards and may be publicly disclosed.)

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ULNRC-05598

ATTACHMENT 1

EVALUATION

EVALUATION

1.0 INTRODUCTION

This amendment application is being submitted for a proposed change to Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," which establishes the program for leakage rate testing of the containment, as required by 10 CFR 50.54, "Conditions of licenses," Section (o) and 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, "Performance-Based Requirements," as modified by approved exemptions, and guided by Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

Specifically, AmerenUE proposes to revise TS 5.5.16 to reflect a one-time five-year deferral of the containment Type A (integrated leak rate) test from once in ten years to once in 15 years. The proposed amendment is risk-informed and follows the guidance in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Bases." AmerenUE has performed an analysis showing that the increase in risk resulting from the proposed amendment is small and within established guidance. AmerenUE has also determined that defense-in-depth principles will be maintained based on risk and other considerations. The proposed change is consistent with Integrated Leak Rate Test (ILRT) interval extensions recently granted to other licensees (i.e., as listed in Section 7.0, "Precedent").

2.0 DESCRIPTION OF PROPOSED TS CHANGE

This application for amendment to the Callaway Technical Specifications proposes to add an exception to the Containment Leakage Rate Testing Program. Specifically, this revision takes a one-time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by RG 1.163 and Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J." The exception allows the performance of a Type A test within fifteen years from the last Type A test, which was performed on October 26, 1999. From a differential safety benefit perspective, the improvement by performing the ILRT within ten years rather than fifteen years is not commensurate with the significant additional cost associated with the test frequency. The specific change to TS 5.5.16, "Containment Leakage Rate Testing Program," is identified below in bold type:

5.5.16 <u>Containment Leakage Rate Testing Program</u>

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exceptions:

    1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

    2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in

accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

3. The unit is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement during Refuel 14 outage (fall of 2005).

4. **The first Type A test performed after the October 26, 1999 Type A test shall be performed no later than October 25, 2014.**

3.0 BACKGROUND

The Callaway Plant containment consists of the concrete containment building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis Loss-of-Coolant Accident (LOCA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a prestressed, reinforced concrete, cylindrical structure with a hemispherical dome and a conventionally reinforced concrete base slab with a central cavity and instrumentation tunnel to house the reactor vessel. A continuous peripheral tendon access gallery below the base slab is provided for the installation and inspection of the vertical post-tensioning system. The base slab, cylinder, and dome are reinforced by bonded reinforcing steel, as required by the design loading conditions. The inside structure is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. A post-tensioning system is used to prestress the cylindrical shell and dome. The system uses unbonded tendons, each consisting of approximately 170 one-quarter-inch-diameter high strength steel wires and anchorage components consisting of stressing washers. The vertical tendons consist of 86 inverted U-shaped tendons, which extend through the full height of the cylindrical wall over the dome and are anchored at the bottom of the base slab. The cylinder circumferential (hoop) tendons consist of 135 tendons anchored at three buttresses equally spaced around the outside of the containment. Prestressing of the hemispherical dome is achieved by a two-way pattern of the inverted U-shaped tendons and 30 hoop tendons, which start at the springline and continue up to an approximate 45-degree vertical angle from the springline.

The concrete containment building is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage-limiting boundary of the containment. Maintaining operability of the containment will limit leakage of fission product radioactivity released from the containment to the environment.

The integrity of the containment penetrations and isolation valves is verified through Type B and Type C Local Leak Rate Tests (LLRTs), and the overall leak tight integrity of the containment is verified by a Type A ILRT, as required by 10 CFR 50, Appendix J.

Option B of Appendix J to 10 CFR 50 requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. Callaway Plant TS 5.5.16 requires that a program be established to comply with the containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. Additionally, this program complies with the guidelines contained in RG 1.163 and NEI 94-01, Revision 0, dated July 26, 1995.

NEI 94-01, Revision 0, specifies an initial test interval of 48 months for Type A tests and allows an extension of the interval to 10 years based on two consecutive successful tests. RG 1.163 endorses NEI 94-01, Revision 0, as a method acceptable to the NRC staff for complying with the performance-based Appendix J, Option B, with four exceptions. Exception 1 discusses the test interval for Type A test. The RG states that ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements," test intervals are not performance-based. Therefore, licensees intending to comply with 10 CFR Part 50 Appendix J, Option B for Type A test intervals must comply with Section 11.0 of NEI 94-01, Revision 0, which refers licensees to Sections 9 and 10 of that document.

NEI 94-01, Revision 0, Section 9.2.3, "Extended Test Intervals," discusses the Type A test. This section states that Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 L_a. Elapsed time between the first and last test in a series shall be at least 24 months.

RG 1.163, Section C, "Regulatory Position," Exception 3 discusses the visual examination of accessible internal and exterior surfaces of the containment system for structural problems. Exception 3 states, "These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration."

The other two exceptions in RG 1.163 are not pertinent to the discussion of Type A test frequencies, as they involve Type B and C testing which is not relevant to this license amendment request.

With the successful completion of the two most recent Type A tests at Callaway Plant and greater than 24 months of elapsed time between those two tests, Callaway Plant has a test interval of every 10 years. The current 10-year interval for the completion of the next Type A test ends on October 25, 2009. Using the 15-month extension provision of NEI 94-01, Revision 0, Callaway Plant currently plans to perform the next Type A test during the spring 2010 refueling outage.

The Callaway Plant containment leakage testing program complies with the requirements of the General Design Criteria and Appendix J to 10 CFR 50. The Type A test is performed at a frequency of once every 10 years based on Type A test performance history. The maximum allowable leakage rate, L_a, at pressure P_a, is 0.200 weight percent (%) per day. For testing purposes, the acceptance criterion is 0.75 L_a or 0.150 weight percent (%) per day. The Type A test is performed in accordance with the provisions of ANSI/ANS 56.8-1994 and NEI 94-01, Revision 0.

Visual inspection of the accessible interior and exterior surfaces of the containment structures and components is performed in accordance with the requirements of American Society of Mechanical Engineers (ASME) Code Section XI, "Inservice Inspection," Subsections IWE, "Requirements for Class NMC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and IWL, "Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants," to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness. If there is evidence of structural deterioration, a Type A test is not performed until all identified irregularities are resolved in accordance with acceptable procedures, nondestructive tests, and inspections.

4.0 TECHNICAL ANALYSIS

4.1 Adoption of 10 CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed allowable leakage rate values specified in the Technical Specification (TS) and TS Bases. The allowable leakage rate is limited such that the leakage assumptions in the safety analyses are not exceeded. The limitation on containment leakage provides assurance that the containment will perform its design function following an accident, up to and including a Design Basis Accident (DBA).

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." For Callaway Plant, the NRC issued License Amendment No. 111 on May 28, 1996 to permit implementation of 10 CFR 50, Appendix J, Option B. Thus, TS 5.5.16 currently requires the establishment of a Containment Leakage Rate Testing Program in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program implements the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," which specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," subject to several regulatory positions stated in RG 1.163.

4.2 TS Changes for Steam Generator Replacement

The NRC issued License Amendment No. 160 on March 17, 2004, for Callaway Plant to modify TS 5.5.16.a as follows:

> "At the end of the paragraph, the following is added to modify compliance with RG 1.1 63: 'as modified by the following exceptions: (1) the visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.
> (2) The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.'"

In the Safety Evaluation Report associated with License Amendment No. 160, the NRC staff agreed with the licensee that the visual examinations of the containment concrete surfaces and the liner plate performed pursuant to Subsections IWL, "Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants," and IWE, "Requirements for Class NMC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," respectively, are more rigorous than those performed pursuant to RG 1.163 and NEI 94-01, Revision 0. The requirements in Subsections IWL and IWE of the American Society of Mechanical Engineers (ASME) Code constitute acceptable requirements for the inspection of the concrete surfaces and the liner plate in the Callaway containment, in that the requirements in Subsections IWL and IWE meet 10 CFR 50.55a, "Codes and standards," Section (b)(2)(vi) and 50.55a(g)(4). Therefore, the NRC staff found that the changes proposed with respect to TS 5.5.16 were acceptable.

The NRC issued License Amendment No. 168 on September 29, 2005, for Callaway Plant to further modify TS 5.5.16.a. as follows:

> "The unit is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement during the Refuel 14 outage (fall of 2005)."

The exception allowed the Callaway Plant to not perform an ILRT after installation of the replacement Steam Generators (SGs) during Refuel 14. The replacement SGs were installed without the containment being cut. During a Loss-of-Coolant Accident (LOCA), portions of the SGs and attached lines are relied on as a barrier against the uncontrolled release of radioactivity to the environment. The portions impacted, that are considered part of the containment boundary, are the outer shell of the SGs, the inside containment portions of lines emanating from the SG shells (i.e., the main steam lines, the main feedwater (MFW) lines, the SG blowdown and sample lines) and the inside surface of the SG tubes.

NEI 94-01, Revision 0, states that either a Type A test (Integrated Leakage Rate Test (ILRT)) or Local Leak Rate Test (LLRT) is required to be conducted prior to returning the containment to operation following a modification that affects containment integrity. Replacing the SGs is such a modification since, as discussed above; the replacement would affect portions of the containment boundary.

However, Callaway Plant stated that performing the ILRT was unnecessary because the ASME Boiler and Pressure Vessel Code (the ASME Code) Section III/XI, "Inservice Inspection," pressure test requirements satisfy the intent of 10 CFR Part 50, Appendix J, Option B. The NRC staff reviewed the ASME Section XI requirements and determined that the ASME Section XI surface examination, volumetric examination, and system pressure testing requirements are more stringent than the ILRT requirements of Appendix J (which are currently stated in TS 5.5.16). Although, the objective of the ILRT is to ensure the leak tight integrity of the containment area affected by the modification, the ASME Section XI inspection and testing requirements more than fulfill the intent of the requirements of Appendix J and the provisions of NEI 94-01, Revision 0. In addition, the test pressure for the ASME Code system pressure test is significantly greater than that of the Appendix J test. Also, the replacement SGs were installed at Callaway without the containment being cut.

In the Safety Evaluation Report associated with License Amendment No. 168, the NRC staff concluded that the licensee's proposed exception from performing a post-modification ILRT following the installation of the replacement SGs was acceptable. The NRC staff further concluded that the containment will continue to meet General Design Criterion (GDC) 50 without conducting the ILRT after the installation of the replacement SGs and, therefore, the proposed change to TS 5.5.16 was acceptable.

4.3 Provisions of Current Program Under Option B

10 CFR 50, Appendix J, Option B, Section V, "Application," specifies that the Regulatory Guide (i.e., RG 1.163) or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TSs. Additionally, deviations from guidelines endorsed in the Regulatory Guide are to be submitted as a revision to the plant's TSs. Therefore, this application for extending the ILRT interval on a one-time basis does not require an exemption from 10CFR 50, Appendix J, Option B, as a license amendment is all that is required.

Adoption of the Option B performance-based containment leakage rate testing program at Callaway Plant did not alter the basic method by which Appendix J leakage rate testing is performed or the associated acceptance criteria, but it did alter the test frequency of Type A, B, and C containment leakage rate tests. The required testing frequency is based upon an evaluation which utilizes the "as found" leakage history to determine the frequency for leakage testing. This provides assurance that leakage will be maintained within required limits.

Such an approach to leak rate testing supports the proposed extension of the ILRT interval for Callaway. Further justification for the proposed change is based on research documented in NUREG-1493, "Performance-Based Containment Leak-Test Program." NUREG-1493 made the following observation with regard to changing the test frequency:

> "Reducing the Type A testing frequency to once per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because Type A tests identify only a few potential leakage paths that cannot be identified by Type B and C testing and the leaks that have been found by Type A tests have only been marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the same fraction of leakage detected solely by Type A testing, increasing the interval between Type A testing had minimal impact on public risk."

4.4 Summary of Test and Inspection Program

Satisfactory results from previous Type A tests at Callaway Plant, as well as continued satisfactory results from Type B and Type C LLRTs and containment inspections, support the proposed one-time extension of the containment Type A test interval.

4.4.1 Integrated Leak Rate Test History

Type A testing is performed to verify the integrity of the containment structure in its LOCA configuration. As stated in NEI 94-01, Revision 0, "The purpose of Type A testing is to verify the leakage integrity of the containment structure. The primary performance objective of the Type A test is not to quantify the overall containment system leakage rate." The Type A testing methodology that is described in ANSI/ANS 56.8-1994 and the modified testing frequencies that are recommended by NEI 94-01, Revision 0, serve to ensure continued leakage integrity of the containment structure. The results of the previous three ILRTs for the containment structure at Callaway Plant, which are presented in Table 4.4.1, validate the leakage integrity of the containment structure.

These historical results indicate that the Callaway Plant containment structure meets the requirements of 10 CFR 50 Appendix J, thus ensuring an essentially leak tight barrier. These plant specific results support the conclusions of NUREG-1493.

Table 4.4.1
Callaway Plant ILRT Results
(% weight per day)

ILRT Test Date	As Found ILRT	Acceptance Limit	As Left ILRT	Acceptance Limit
April 28, 1987	0.0830	≤ 0.2000	0.0440[1]	≤ 0.1500
October 30, 1990	0.1987	≤ 0.2000	0.0524[2]	≤ 0.1500
October 26, 1999	0.0533	≤ 0.2000	0.0445[2]	≤ 0.1500

[1] Total Time Calculated Upper 95% Confidence Leakage Rate
[2] Mass Point Calculated Upper 95% Confidence Leakage Rate

Leakage rates, in the table above, are expressed in units of containment air weight percent per day at a test pressure equal to the calculated peak containment internal pressure for the limiting DBA (48.1 psig). Calculated as-left results are expressed at a 95% confidence level plus leakage attributed to non-vented penetrations and corrections for volume changes. Calculated as-found results are expressed at a 95% confidence level plus leakage attributed to non-vented penetrations plus leakage improvements made to the containment boundary during the outage prior to conducting the ILRT. The maximum allowable as-left containment leakage rate allowed by Option B during containment leak rate testing was 0.1500 weight percent per day (0.75 L_a). The maximum allowable as-found containment leakage rate allowed by Option B during containment leak rate testing was 0.2000 weight percent per day (1.0 L_a).

The as-left ILRT results from 1987 to 1999 are very consistent within the expected variability of test methods, test techniques, and plant conditions. The as-found ILRT result includes leakage savings, i.e., the difference between the as-found minimum pathway leakage and the as-left minimum pathway leakage for each Type B and C penetration tested and adjusted prior to the ILRT. The unusually large 1990 as-found ILRT result was primarily due to large leakage savings for two containment penetrations. Penetration P-71 had an as-found minimum pathway LLRT result of 117,380 sccm, which after repair and retest, was reduced to an as-left minimum pathway LLRT result of 594.4 sccm. Similarly, Penetration P-73 had an as-found minimum pathway LLRT result of 185,760 sccm, which after repair and retest, was reduced to an as-left minimum pathway LLRT result of 282.4 sccm. Since the adoption of Option B for Callaway Plant in 1996, these two penetrations have not exceeded their administrative leakage limit. The low as-found 1999 ILRT result indicates that these penetrations have been successfully addressed in the Callaway Plant Type B and C testing program.

As discussed, in NEI 94-01, Revision 0, Section 9.2.5, total leakage savings are identified through the performance of Type B and Type C testing and do not contribute significantly to performance of a Type A (ILRT) test. Since there were no penetrations isolated due to excessive leakage during the performance of the 1999 ILRT, the 1999 Performance Leakage Rate was calculated to be 0.0445% weight per day. This is well below the acceptance criteria of 0.2% weight per day and the Callaway Plant ILRT frequency qualified to remain on a ten-year frequency.

4.4.2 Type B and C Testing

Type B and C testing at Callaway Plant ensures that containment penetrations such as air locks, flanges, sealing mechanisms and containment isolation valves are essentially leak tight. There are no pressure-retaining bellows used on containment penetrations at Callaway Plant.

The initial test frequency for performing a leak test on Type B and Type C components is a base interval of 30 months. For Type B components, the interval may be extended to up to 120 months based on acceptable performance. Type B components whose test intervals are extended to greater than 60 months are tested on a staggered basis to allow for early detection of any common mode failure mechanism. For Type C components, the interval may be extended up to 60 months based upon acceptable performance. Acceptable performance for extending the 30-month interval is established by passing two as-found LLRTs with leakage less than or equal to the established administrative limits and that are at least 24 months apart (or separated by a normal refueling interval).

In accordance with TS 5.5.16, Type B and C acceptance criteria prior to unit startup must be less than 0.60 L_a. Table 4.4.2a provides a summary of the as-left maximum pathway (MXPLR) running total for Type B and C components from Refuel (RF) 13 (Spring 2004) to the most recent refueling outage, Refuel

16 (Fall 2008). As illustrated in the table, the as-left Type B and C maximum pathway leakage totals have been well below the acceptance criterion of 0.60 L_a (0.12% weight per day).

Table 4.4.2a
Callaway Plant LLRT Results
(% weight per day)

Refuel Outage No.	Date	MXPLR	Acceptance Criterion
13	Spring 2004	0.039	< 0.12
14	Fall 2005	0.041	< 0.12
15	Spring 2007	0.025	< 0.12
16	Fall 2008	0.068	< 0.12

For the refueling outages listed above, Table 4.4.2b provides a summary list of those Type B and C containment components that have not demonstrated acceptable performance history in accordance with the primary containment leakage rate program. Components that have not demonstrated acceptable performance history are on a 30 month test frequency in accordance with the requirements of NEI 94-01, Revision 0.

Table 4.4.2b
Callaway Plant Components
Exceeding Administrative Limits

Component ID	Date	Measured Leakage (sccm)	Administrative Limit (sccm)	Corrective Action
EMV0006	Spring 2007	6,658	5,000	Valve disc was replaced. Valve was retested in most recent outage (Refuel 16) with a measured leakage of 3,100 sccm.
EFHV0046/ EFHV0048/ EFHV0050	Fall 2008	46,400	34,000	Valves are tested together. Corrective action for Valves EFHV0048 and EFHV0050 is scheduled for Refuel 17 (Spring 2010).
KCV0478	Fall 2008	13,800	10,000	Internals of check valve were cleaned. As-left leakage rate was 645 sccm.

Administrative leakage limits are conservatively assigned to each Type B and Type C component as an indicator of potential degradation and are not surveillance acceptance criteria. Failure to meet an administrative leakage limit is <u>not</u> a failure of the component. The administrative leakage limits are set at a low value to allow for early identification of potential degradation and scheduling of corrective action.

Table 4.4.2c provides the schedule for the Type B and Type C tests on containment pressure-retaining boundaries that are or will be scheduled to be performed prior to and during the requested one-time 5-year extension period of the ILRT interval. The due date column provides the date of the next scheduled

LLRT for the penetration. For Callaway Plant LLRTs, the due date is established based on the last completed date plus the frequency.

Table 4.4.2c Callaway Plant LLRT Schedule				
Callaway Plant Penetration Number	**Valve Number**	**Last Completed**	**Due Date**	**Frequency**
P-017	NA	RF16	RF17	18 MONTH
[redacted]				
P-022	BBHV8351B	RF15	RF18	54 MONTH
	BBV0148	RF14	RF17	54 MONTH
[redacted]				
P-023	BGHV8152	RF15	RF18	54 MONTH
	BGHV8160	RF15	RF18	54 MONTH
[redacted]				
P-024	BGHV8112	RF14	RF17	54 MONTH
	BGHV8100	RF15	RF18	54 MONTH
	BGV0135	RF14	RF17	54 MONTH
[redacted]				
P-025	BLHV8047	RF15	RF18	54 MONTH
	BL8046	RF14	RF17	54 MONTH
[redacted]				
P-026	HBHV7136	RF15	RF17	36 MONTH
	HBHV7176	RF15	RF17	36 MONTH
[redacted]				
P-028	EFHV0032	RF16	RF18	36 MONTH
	EFHV0034	RF15	RF17	36 MONTH
[redacted]				
P-029	EFHV0046	RF16	RF17	18 MONTH
	EFHV0048/50	RF16	RF17	18 MONTH
[redacted]				
P-030	KAFV0029	RF15	RF18	54 MONTH
	KAV0204	RF14	RF17	54 MONTH
	KAV0218	RF14	RF17	54 MONTH
[redacted]				
P-032	LFFV0095	RF15	RF18	54 MONTH
	LFFV0096	RF15	RF18	54 MONTH
[redacted]				
P-034	NA	RF16	RF17	18 MONTH
[redacted]				
P-036	NA	RF16	RF17	18 MONTH
[redacted]				
P-039	BBHV8351C	RF15	RF18	54 MONTH
	BBV0178	RF14	RF17	54 MONTH
[redacted]				
P-040	BBHV8351D	RF15	RF18	54 MONTH
	BBV0208	RF14	RF17	54 MONTH
[redacted]				
P-041	BBHV8351A	RF15	RF18	54 MONTH

Table 4.4.2c
Callaway Plant LLRT Schedule

Callaway Plant Penetration Number	Valve Number	Last Completed	Due Date	Frequency
	BBV0118	RF14	RF17	54 MONTH
P-043	HDV0016	RF16	RF19	54 MONTH
	HDV0017	RF16	RF19	54 MONTH
P-044	HBHV7126	RF15	RF18	54 MONTH
	HBHV7150	RF15	RF18	54 MONTH
P-045	EPV0046	RF15	RF17	36 MONTH
	EPHV8880	RF15	RF17	36 MONTH
P-050	NA	RF16	RF17	18 MONTH
P-051	Blind Flange	RF16	RF19	54 MONTH
	Blind Flange	RF16	RF19	54 MONTH
P-053	ECV0083	RF15	RF18	54 MONTH
	ECV0084	RF16	RF19	54 MONTH
P-054	ECV0087	RF16	RF19	54 MONTH
	ECV0088	RF16	RF19	54 MONTH
P-055	ECV0095	RF16	RF19	54 MONTH
	ECV0096	RF16	RF19	54 MONTH
P-056	GSHV0008	RF14	RF17	54 MONTH
	GSHV0009	RF14	RF17	54 MONTH
	GSHV0038	RF14	RF17	54 MONTH
	GSHV0039	RF14	RF17	54 MONTH
P-058	EMV0006	RF16	RF17	18 MONTH
	EMHV8888	RF16	RF17	18 MONTH
P-062	BBHV8026	RF14	RF17	54 MONTH
	BBHV8027	RF14	RF17	54 MONTH
P-063	KAV0039	RF14	RF17	54 MONTH
	KAV0118	RF14	RF17	54 MONTH
P-064	SJHV0129/130	RF14	RF17	54 MONTH
	SJHV0128	RF14	RF17	54 MONTH
P-065	GSHV0020	RF14	RF17	54 MONTH
	GSHV0021	RF14	RF17	54 MONTH

Table 4.4.2c
Callaway Plant LLRT Schedule

Callaway Plant Penetration Number	Valve Number	Last Completed	Due Date	Frequency
P-067	KCV0478	RF16	RF17	18 MONTH
	KCHV0253	RF15	RF17	18 MONTH
P-068	NA	RF16	RF17	18 MONTH
P-069	SJHV0012	RF14	RF17	54 MONTH
	SJHV0013	RF14	RF17	54 MONTH
P-071	EFHV0031	RF16	RF17	18 MONTH
	EFHV0033	RF15	RF17	36 MONTH
P-073	EFHV0045	RF15	RF17	36 MONTH
	EFHV0047/49	RF15	RF17	36 MONTH
P-074	EGV0204	RF16	RF17	18 MONTH
P-074 cont.	EGHV0058/127	RF16	RF17	18 MONTH
P-075	EGHV0060/130	RF14	RF17	54 MONTH
	EGHV0059/131	RF14	RF17	54 MONTH
P-076	EGHV0061/133	RF14	RF17	54 MONTH
	EGHV0062/132	RF14	RF17	54 MONTH
P-078	BMV0045	RF14	RF17	54 MONTH
	BMV0046	RF14	RF17	54 MONTH
P-080	BG8381	RF14	RF17	54 MONTH
	BGHV8105	RF14	RF17	54 MONTH
P-092	EMHV8871	RF15	RF18	54 MONTH
	EMHV8964	RF15	RF18	54 MONTH
P-093	SJHV0005	RF15	RF18	54 MONTH
	SJHV0006/127	RF15	RF18	54 MONTH
P-095	SJHV0018	RF14	RF17	54 MONTH
	SJHV0019	RF14	RF17	54 MONTH
P-097	GSHV0017	RF16	RF18	36 MONTH
	GSHV0018	RF16	RF18	36 MONTH
	GSHV0033	RF16	RF18	36 MONTH
	GSHV0034	RF16	RF18	36 MONTH
P-098	KBV0001	RF15	RF18	54 MONTH
	KBV0002	RF15	RF18	54 MONTH

Table 4.4.2c
Callaway Plant LLRT Schedule

Callaway Plant Penetration Number	Valve Number	Last Completed	Due Date	Frequency
P-099	GSHV0003	RF15	RF17	36 MONTH
	GSHV0004	RF15	RF17	36 MONTH
	GSHV0005	RF15	RF17	36 MONTH
	GSHV0036	RF15	RF17	36 MONTH
	GSHV0037	RF15	RF17	36 MONTH
P-101	GSHV0012	RF15	RF17	36 MONTH
	GSHV0013	RF15	RF17	36 MONTH
	GSHV0014	RF15	RF17	36 MONTH
	GSHV0031	RF15	RF17	36 MONTH
	GSHV0032	RF15	RF17	36 MONTH
P-160	GTHZ0008	Note 1	Note 1	92 DAYS
	GTHZ0009	Note 1	Note 1	92 DAYS
P-160 cont.	GTHZ0011/12	Note 1	Note 1	92 DAYS
P-161	GTHZ0004/5	Note 1	Note 1	92 DAYS
	GTHZ0006	Note 1	Note 1	92 DAYS
	GTHZ0007	Note 1	Note 1	92 DAYS
L-001	Barrel	RF16	RF17	30 MONTH
L-002	NA	RF16	RF17	18 MONTH
L-003	Barrel	RF16	RF17	30 MONTH
ELECTRICAL PENETRATIONS	N/A	RF10	RF17	10 YEAR

Refueling Outage Schedule

RF13 – Spring 2004
RF14 – Fall 2005
RF15 – Spring 2007
RF16 – Fall 2008
RF17 – Spring 2010
RF18 – Fall 2011
RF19 – Spring 2013

Note 1 – In accordance with Technical Specification Requirements, these valves are tested every 92 days.

The current Type B and C penetration test frequencies are established based on performance using the requirements of 10 CFR 50, Appendix J, Option B. The test frequencies are re-evaluated after each refueling outage for potential changes. For example, if a containment boundary component that is on an extended testing frequency fails to meet its administrative limit, its frequency will be increased to at least once every 30 months. Likewise, if a containment boundary component that is not on an extended testing frequency meets the performance criteria of NEI 94-01, Revision 0, its frequency may be extended. Planned maintenance, valve diagnostic testing, and other plant activities may affect the frequencies listed in the table above. The Type B and C testing requirements will not be changed as a result of the proposed license amendment.

4.4.3 Containment Inspections

4.4.3.1 Visual Inspections

Prior to issuance of Callaway Plant Amendment No. 160 (March 17, 2004), as part of the Appendix J Program, Callaway Plant performed visual inspections of accessible interior and exterior surfaces of the containment system for structural problems that may have affected either the containment structural leakage integrity or performance of the Type A Test. These examinations were conducted in accordance with the requirements of 10 CFR 50 Appendix J, RG 1.163, and NEI 94-01, Revision 0, prior to any Type A test, to uncover any evidence of structural deterioration which may have affected either the containment structural integrity or leak tightness, as well as during the two refueling outages conducted before the next Type A test, based on a ten-year frequency.

Subsequent to Amendment No. 160, additional visual inspections have been conducted in accordance with the requirements of ASME Section XI, Subsection IWE (ASME IWE) and Subsection IWL (ASME IWL). The results of these inspections of accessible interior and exterior surfaces of the containment system are detailed in Section 4.4.3.2 and 4.4.3.3 below.

Since the last Type A test in 1999, an Appendix J visual inspection of the accessible interior and exterior surfaces of the Callaway Plant containment was completed on May 2, 2001. This visual inspection indicated that that there were no structural problems that could have affected the containment structural leakage integrity. Pursuant to License Amendment No. 160, and as noted above, additional visual inspections were conducted under ASME Subsections IWE and IWL and are discussed in the next section.

The more rigorous visual examinations of the containment concrete surfaces and the liner plate performed pursuant to Subsections IWL and IWE, in accordance with License Amendment No. 160 requirements, will not be changed as a result of the proposed change.

4.4.3.2 IWE Containment Inservice Inspection (CISI) Program

A comprehensive containment pressure boundary and related components inspection is performed at Callaway Plant in accordance with the requirements of ASME Section XI Subsection IWE. The acceptance criteria used for the examination of IWE components are established by AmerenUE and comply with Subsection IWE-3000 of the ASME code.

The Callaway Plant IWE Containment Inservice Inspection (CISI) Program was initially developed in accordance with the 1992 Edition of the ASME Boiler and Pressure Vessel Code, Section XI. Prior to performance of the initial inspection, Callaway Plant submitted a relief request (Relief Request ULNRC-03938) and received approval to use the 1998 Edition of the ASME Code, Section XI Subsection IWE as supplemented by specific commitments documented in letters to the NRC, i.e., ULNRC-04063 and ULNRC-04109.

Inspections covering the requested five-year extension period (2009-2014) will be in accordance with 10CFR50.55a, including successive 120-month interval updates in accordance with paragraph (g)(4)(ii). This paragraph states that, "Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month inspection interval (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147 through Revision 15, that are incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section." There are no known anticipated Relief Requests for the next Program Update.

The initial inspections and tests (i.e., first-period ASME IWE examinations) were completed at Callaway Plant prior to September 2001. Since 2001, inspections and tests have been completed in 2002, 2005, and 2008 in accordance with the frequency specified in ASME IWE. ASME IWE allows inaccessible areas, up to 20% of the total area of the pressure boundary, to be exempted from inspection. Callaway Plant has 16% of the total pressure boundary that is considered inaccessible due to concrete, insulation, or other obstructions that prevent visual access to the area. As long as there is no evidence of deterioration on the areas adjacent to the inaccessible areas, further inspections are not required. During the 1999, 2002, 2005 and 2008 IWE inspections, AmerenUE identified various indications that were either repaired or documented and evaluated as acceptable by the Responsible Engineer with no loss of containment leak-tightness. Indications and findings are discussed below.

<u>Initial IWE Inspection</u>

During Refuel 10, in October 1999, Callaway Plant performed the initial IWE inspection. The inspection performed during Refuel 10 included 100% of the accessible areas of the Class MC components and the metallic liner of Class CC components of the containment pressure boundary for deformation caused by heavy impact, bulges, wrinkles, large cracks, excessive corrosion or any signs of deterioration. In addition, all bolted connections of the pressure boundary that were disassembled during Refuel 10 had VT-1 inspections on bolting materials, torque, and gasket surfaces. The general condition of the Callaway Plant Pressure Boundary Liner Plate was good with only a few limited areas with minor problems. The following areas of concern were identified during the inspection: the incore tunnel sump, containment normal sumps "A" and "B", recirculation sump "A", spare electrical penetration E286, containment dome liner, and the RHR and Containment Spray encapsulation domes. These areas were entered into the Callaway Plant corrective action program and are discussed below.

The incore and containment normal sumps experienced some general corrosion and corrosion pitting. Although the corrosion pitting was generally in the 1/16 inch depth or less range, there were some indications of 1/8 inch depth. Due to identifying the early stages of corrosion and deterioration in the containment normal sumps ("A" and "B"), as identified during the Refuel 10 IWE inspections, new 1/4-inch stainless steel liner plates were installed over the existing carbon steel liner during Refuel 11 ("B" sump) and Refuel 12 ("A" sump). The stainless steel liner plate is now the ASME Section XI, IWE pressure boundary.

The indication in recirculation sump "A" was approximately 3/8 inch in diameter and 3/64 inch in depth and appeared to be the result of impact from a tool or heavy object and not the result of deterioration of the coating. Similarly, an indication 3/8 inch long and of less than 1/8 inch depth on the end cap of spare electrical penetration E286 appeared to be the result of impact from a falling object. There was no general corrosion around the indication and no other signs of distress. These non-conformances were evaluated using the Callaway Plant corrective action program and found to be acceptable for continued use. Work requests were initiated to clean and paint the areas of the liner plate affected by corrosion.

An arc strike measuring 2-1/2 inches long and less than 1/32 inch in depth was found on the containment dome liner. The arc strike had been on the liner since construction and was not the result of any recent activity. The arc strike was within the acceptance criteria such that further action was not warranted. Two encapsulation domes were disassembled during Refuel 10, and four minor scratches were discovered on lower flange of one of the domes. The scratches were evaluated and found acceptable.

First Period IWE Inspection

During Refuel 12, in October 2002, Callaway Plant performed the first period inspection of the containment pressure boundary. The inspection examined 100% of accessible areas of the Class MC components and the metallic liner of Class CC component of the containment pressure boundary for deformation caused by heavy impact, bulges, wrinkles, large cracks, excessive corrosion or any signs of deterioration. In addition, all bolted connections of the pressure boundary that were disassembled during Refuel 12, had VT-1 inspections on bolting materials, torque, and gasket surfaces. The general condition of the Callaway Plant containment pressure boundary was considered good with no major areas of concern and only a few minor areas of corrosion pitting and indications were identified. A follow-up, on the areas of concern identified during the initial Refuel 10 inspection, indicated that only completion of painting in the incore pit remained outstanding. There were five (5) areas of concern that were identified during the inspection. These areas were entered into the Callaway Plant corrective action program and are discussed below.

The five areas of concern were unattached welds on the liner plate, containment liner plate rust, containment normal sump "A" corrosion, unacceptable 3/4-inch nuts found during VT-1 exam, and containment liner paint damage. The unattached welds were from plant construction and had been used for attachment of rigging devices to position the liner plate sections into place. Upon completion of the liner plate, the rigging attachments were removed and the welds were not ground flush with the liner plate. The unattached welds were evaluated and found to be acceptable.

Upon additional examination, the reported rust on the liner plate was determined to be light concrete splatter from plant construction that had discolored and accumulated dust over time. There was no pitting or corrosion of the liner observed. An additional area was also examined and found to be a discoloration, most likely from an oil-like compound coming in contact with the liner plate. These areas were evaluated and found to be acceptable.

The inspection of the containment normal sump "A" liner plate occurred before the new stainless steel liner was installed over the existing carbon steel liner plate. Implementation of the stainless steel liner in containment normal sump "A" was completed during Refuel 12 and is acceptable for use.

Four (4) 3/4-inch nuts on the "B" encapsulation tank failed the acceptance criteria during the VT-1 exam and were removed during Refuel 12. The nuts were replaced under the ASME Section XI repair/replacement plan.

Minor nicks and paint damage were identified along the base of the liner plate on the 2000 foot elevation at the interface of the concrete floor and the liner. The areas did not exceed Callaway Plant acceptance criteria, but repairs were scheduled for Refuel 13 as planned preventative maintenance.

Second Period IWE Inspection

During Refuel 14, in October 2005, Callaway Plant performed the second period inspection of the containment pressure boundary. The inspection examined 100% of accessible areas of the Class MC components and the metallic liner of Class CC component of the containment pressure boundary for deformation caused by heavy impact, bulges, wrinkles, large cracks, excessive corrosion or any signs of deterioration. In addition, all bolted connections of the pressure boundary that were disassembled during Refuel 14 had VT-1 inspections on bolting materials, torque, and gasket surfaces. The general condition of the Callaway Plant containment pressure boundary was considered good with no major areas of concern and only one minor area of corrosion pitting was identified. The one area of concern was entered into the Callaway Plant corrective action program and is discussed below.

Three isolated areas (each approximately 1.5 square feet) of corrosion growth on the incore tunnel liner plate were identified with a maximum depth of pitting of 0.020 inches. This is below the acceptance criteria of 0.025 inches. In addition, corrosion growth (approximately 4 to 6 feet in length) had re-formed at the intersection of the incore tunnel floor and the liner plate wall. As a preventative measure, these areas were added to a Refuel 14 work order to clean and coat these areas.

Third Period IWE Inspection

During Refuel 16, in October 2008, Callaway Plant performed the third period inspection of the containment pressure boundary. The inspection examined 100% of accessible areas of the Class MC components and the metallic liner of Class CC component of the containment pressure boundary for deformation caused by heavy impact, bulges, wrinkles, large cracks, excessive corrosion or any signs of deterioration. In addition, all bolted connections of the pressure boundary that were disassembled during Refuel 16 had VT-1 inspections on bolting materials, torque, and gasket surfaces. The general condition of the Callaway Plant containment pressure boundary was considered good with no major areas of concern. There were three (3) minor areas of concern that were identified during the inspection. These areas were entered into the Callaway Plant corrective action program and are discussed below.

(1) Rust was identified around portions of the Personnel and Emergency Personnel Hatches during the inspection. The area around the Personnel Hatch contained only intermittent surface rust with no internal corrosion. The area was free of blistered, flaked and peeled paint and had no bulges, deformation or other areas of distress, and therefore was determined to be acceptable. The rust area around the Emergency Personnel Hatch was more extensive. As a result, the area was cleaned and repaired.

(2) A leak at the weld area in the lower back right corner of containment normal sump "B" was identified during the inspection and appeared to be from a weld flaw from behind the stainless steel plate. In addition, there was general corrosion and pitting identified. As the general corrosion observed was in small areas throughout the sump and the pitting in these areas was less than 1/16 inch in depth, this condition was acceptable. Containment normal sump "A" was then inspected with no problems identified.

(3) During Refuel 11, and as noted previously, containment normal sump "B" was relined with 1/4-inch stainless steel over the original 1/4-inch carbon steel plate. When the new liner was installed, the only noticeable corrosion on the existing liner plate was on the sump plate surface at the water line and was estimated to be approximately 10% of the thickness. The remaining area of the sump had the inorganic zinc coating intact and had no visible corrosion. The new liner was placed in direct contact with the original liner without compromising the integrity of the original

liner. This would allow the original liner to continue to function as a pressure boundary. Although the original liner cannot be directly observed, any additional corrosion could be detected by a bulging in the stainless steel liner due to rust occurring in a confined space between the two liners and by in-seepage of oxygenated water. An Operability Determination was completed during Refuel 16, which determined that the containment pressure boundary remained intact. No sign of bulging was observed in the sump and stains observed at the weld flaw did not appear to be oxygenated water (rust). Using methodology outlined in Crane Technical Paper 410, and an assumed through-wall flaw of 1/8 inch diameter, a leak rate of 2.99 scfm was calculated. The available margin based on the last ILRT and using only 75% of the maximum allowable leakage rate was 8.07 scfm. Thus, the calculated leakage rate is well below the available margin. A weld repair has been planned for Refuel 17 (Spring 2010).

Based on inspections of accessible areas, described above, a low potential exists for degradation in inaccessible areas. The IWE program examinations have demonstrated that the leak-tightness of the Callaway Plant containment has not been compromised and thus, Callaway Plant does not have any required augmented inspections. The last completed IWE inspection was performed during Refuel 16 (October 2008) and the next scheduled IWE inspections are Fall 2011 and Fall 2014. There will be no change to the CISI schedule at Callaway Plant as a result of the proposed change. AmerenUE will conduct CISI inspections of the containment at Callaway Plant, as originally scheduled per the CISI Program.

4.4.3.3 IWL Containment Inservice Inspection (CISI) Program

A comprehensive containment exterior concrete and tendon inspection is performed at Callaway Plant in accordance with the requirements of ASME Section XI Subsection IWL. The acceptance criteria used for the examination of IWL components are established by AmerenUE and comply with Subsection IWL-3000 of the ASME code. The tendon surveillance program consists of periodic inspection of the condition of a randomly selected group of tendons. Physical tendon surveillance consists of sheathing filler inspection, anchorage inspection, tendon lift-off, inspection and tensile test of removed wire samples (for detensioned tendons) and tendon retensioning with the tendons being resealed after completion of all inspections.

The Callaway Plant IWL Containment Inservice Inspection (CISI) Program was developed in accordance with the 1992 Edition, 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI. Callaway Plant submitted a relief request (via letter ULNRC-03934) and received approval to extend the maximum direct examination distance and reduce the minimum illumination requirements specified in Table IWA-2210-1 when performing concrete surface inspections remotely as required by Subsection IWL, Paragraph IWL-2510.

Inspections covering the requested five-year extension period (2009-2014) will be in accordance with 10CFR50.55a including successive 120-month interval updates in accordance with paragraph (g)(4)(ii). This paragraph states that, "Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month inspection interval (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147 through Revision 15, that are incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section." There are no known anticipated Relief Requests for the next Program Update.

The initial inspections and tests (i.e., first period ASME IWL examinations) were completed at Callaway Plant prior to September 2001. Since 2001, inspections and tests have been completed in accordance with the frequency specified in ASME IWL. During the 1999 and 2004 IWL inspections, AmerenUE identified various indications that were either repaired, or documented and evaluated as acceptable by the Responsible Engineer with no loss of containment structural integrity. Indications and findings are discussed below.

First Period IWL Inspection

During Refuel 10, in October 1999, Callaway Plant performed the first period IWL inspection. The inspection included the containment concrete and the containment post-tensioning system. A total of six (6) tendons (three vertical and three horizontal) were randomly selected and examined during this inspection. One vertical and one horizontal tendon were selected for detensioning. The lift-off forces for these tendons were within or above the predicted limits. Visual inspection of the anchorage system and a wire removed from one of the tendons revealed proper coverage by the filler material with no signs of corrosion or presence of water. No significant structural deterioration was noted during a general exterior visual inspection of the exposed accessible concrete surfaces of the containment. There was one area of concern that was identified during the tendon inspection. This area was entered into the Callaway Plant corrective action program and is discussed below.

During re-greasing of containment tendons 40BC and 6AC, the amount of grease replaced was in excess of 5% which is the maximum allowed by FSAR Chapter 16, Section 16.6.1.2.1.a.(d).2. The measured grease voids were 6.0% (40BC) and 5.7% (6AC). Since the void was discovered after filling, FSAR action statement 16.6.1.2.b (restoration of the required level of integrity) was immediately satisfied with the exception of the Special Report provision. As required by the FSAR, an engineering evaluation was performed and Special Report 99-01, dated June 28, 1999, was sent to the NRC. Previous inspection results indicate that grease voids do not affect the corrosion inhibiting properties of the grease. From tensile tests on the removed wires and from visual inspections of the filler grease and tendon components, the filler grease is performing its intended function of prohibiting or arresting corrosion of the tendons. Based on lift-off results, visual inspections and results from previous inspections, the structural integrity of the tendon, tendon anchorage system and containment building have not been adversely affected by the measured voids.

Second Period IWL Inspection

Callaway Plant performed the second period IWL inspection during the period of July 2004 to September 2004. The inspection included the containment concrete and the containment post-tensioning system. A total of six (6) tendons (three vertical and three horizontal) were randomly selected and examined during this inspection. One vertical and one horizontal tendon were selected for detensioning. The lift-off forces for these tendons were above the predicted limits.

A comparison of the lift-off forces during this inspection to the original installation lock-off forces was made to detect any evidence of system degradation. The comparison revealed that the average tendon losses were 7.66% for the vertical tendons, 11.31% for the horizontal tendons, and 7.56% for the hoop tendons above the spring line. These results were within the expected losses of 8 to 13% for vertical tendons and 9 to 14% for horizontal tendons. A regression analysis was performed on each of the tendon groups, which showed that each group remains above the minimum requirements. Projections showed the vertical tendons, at that time, to be at 1396 kips with a minimum requirement of 1160 kips and horizontal tendons, at that time, to be at 1315 kips with a minimum requirement of 1228 kips.

Grease samples were tested and found to have acceptable levels of water soluble ions and water content. None of the inspected tendons exhibited any presence of water in the grease either during removal of the grease can, detensioning, or around the tendon anchorage at any time. Acceptable corrosion levels were found at all tendon ends and no cracks were found on any anchorage component. All of the bearing plates were found with acceptable levels of corrosion. Concrete cracks surrounding the bearing plates were within allowable tolerance of ≤ 0.010 inch except for one tendon where a 0.030 inch crack was previously reported and the condition had not changed. No additional broken, missing or protruding wires were found on any of the inspected tendons. All grease can inspections were acceptable. All wire samples tested were acceptable in diameter, yield strength and ultimate strength.

No significant structural deterioration was noted during a general exterior visual inspection of the exposed accessible concrete surfaces of the containment. There was an area of concern that was identified during the tendon inspection. Four of the six tendons had grease voids in excess of 5%. These four occurrences were entered into the Callaway Plant corrective action program and are discussed below.

During re-greasing of containment tendons 44BC, 48AC, V13 and V30, the amount of grease replaced was in excess of 5% which is the maximum allowed by FSAR Chapter 16, Section 16.6.1.2.1.a.(d).2. The measured grease voids were 12.8% (44BC), 5.81% (48AC), 6.91% (V13) and 9.76% (V130). Since a void was discovered after filling, FSAR action statement 16.6.1.2 b (restoration of the required level of integrity) was immediately satisfied with the exception of the Special Report provision. As required by the FSAR, an engineering evaluation was performed and Special Report ULNRC-05053, dated September 17, 2004, was sent to the NRC. Previous inspection results indicate that grease voids do not affect the corrosion inhibiting properties of the grease. From tensile tests on the removed wires and from visual inspections of the filler grease and tendon components, the filler grease is performing its intended function of prohibiting or arresting corrosion of the tendons. Based on lift-off results, visual inspections and results from previous inspections, the structural integrity of the tendon, tendon anchorage system and containment building have not been adversely affected by the measured voids.

Based on inspections of accessible areas, described above, a low potential exists for degradation in inaccessible areas. The IWL program examinations have demonstrated that the structural integrity of the Callaway Plant containment has not been compromised and thus, Callaway Plant does not have any required augmented inspections. The last completed IWL inspection was September 15, 2004 and the next scheduled inspections are June 20, 2009 and June 20, 2014. There will be no change to the CISI schedule at Callaway Plant as a result of the proposed change. AmerenUE will conduct CISI inspections of the containment at Callaway Plant, as originally scheduled per the CISI Program.

4.4.3.4 Coatings Inspections

AmerenUE conducts periodic inspections of Service Level 1 coatings inside containment during each refueling outage at Callaway Plant, as required by the plant licensing basis and station procedures. These inspections meet the requirements of RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." As localized areas of degraded coatings are identified, those areas are evaluated and scheduled for repair or replacement, as necessary. The results of recent inspections of containment coatings at Callaway Plant are summarized below. Areas of concern were entered into the Callaway Plant corrective action program.

Refuel 12 (Fall 2002)

Paint peeling from an area of approximately 3 feet by 3 feet on the containment liner was identified. The most likely cause was either insufficient surface preparation or failure of the coating system. A walkdown of all elevations in containment identified five additional areas that had evidence of Carboline 890 not bonding to Carboline 890 or Amercoat 90. The areas were generally small and would not have significantly impacted the flow of water to the recirculation sump. The areas were repaired and re-coated.

As a result of the Carboline 890 paint failures identified in Refuel 12, a root cause investigation was performed to prevent similar failures in the future. Carboline 890 is widely used within the nuclear power industry as a qualified coating system. The failures identified in Refuel 12 were primarily due to inadequate surface preparation. Callaway Plant had been using a solvent wipe surface preparation process. Procedures were changed to incorporate a more aggressive hand and/or power tool cleaning surface preparation process and stricter controls on the applied film thickness of the Carboline 890 coating system.

Refuel 13 (Spring 2004)

Two areas of paint flaking were identified during a walkdown of containment coatings. These areas had previously been identified during Refuel 12 and had been scheduled for repair. The repair was completed during Refuel 13.

Refuel 14 (Fall 2005)

The overall condition of the coatings was good with only the following minor corrective actions: 1) Cleaning and re-coating of the floor to wall intersection in the incore tunnel, 2) Repair of three isolated areas in the incore tunnel liner, 3) Repair of damaged coating on the wall between the "A" and "B" steam generators, and 4) Removal of duct tape from containment wall near azimuth 242 degrees, elevation 2047 feet.

Refuel 15 (Spring 2007)

Several areas of damaged coatings were observed at various areas of containment. Each identified area was small and it was conservatively estimated that the total surface area of damaged coatings identified was seven (7) square feet. A work order was initiated to remove the damaged areas and repair the coatings.

Refuel 16 (Fall 2008)

Several areas of damaged coatings were observed at various areas of containment. Each identified area was small and it was conservatively estimated that the total surface area of damaged coatings identified was seven (7) square feet. A work order was initiated to repair the damaged areas.

Paint flaking and peeling was identified at nine zones on the containment liner during the ASME Section XI IWE Containment Pressure Boundary Inspection. A sump blockage evaluation was performed that concluded there was no adverse effect on the ability of the ECCS to perform as designed. The areas were repaired during Refuel 16.

There were no imminent concerns of coating deterioration that would affect the safe operation or safe shutdown of the plant. The inspection requirements of the containment coatings program at Callaway Plant will not be changed as a result of the proposed changes, including scheduled coating inspections for the upcoming refueling outages.

4.4.3.5 Maintenance Rule Inspections

Maintenance Rule Baseline Inspections required by 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," were completed for Callaway Plant. The inspections included the internal containment structures. At Callaway Plant, structures including the containment building are monitored within the scope of the Maintenance Rule. This monitoring takes credit for the IWE and IWL inspections and is not a separate inspection. The Maintenance Rule program does not require any additional inspections of pressure-retaining SSCs of the containment system that come under the purview of the Appendix J Containment Leakage Testing Program.

Based upon these baseline inspections, AmerenUE concluded that these structures are being adequately maintained and capable of performing their intended functions. This program ensures that internal containment structures at Callaway Plant are evaluated and maintained in a condition to perform their intended functions. There have been no corrective actions initiated as a result of Maintenance Rule inspections at Callaway Plant. There will be no changes to the Maintenance Rule Program as a result of the proposed changes.

4.5 Risk Assessments

An assessment was performed to evaluate the risk impact of reducing the currently allowed containment ILRT (Type A) frequency from one test in 10 years to one test in 15 years for Callaway Plant (Attachment 4) on a one-time basis. As a result, the proposed extension was found to have a small increase in risk while it would allow for substantial cost savings. The proposed change would only impact testing associated with the current surveillance test for Type A leakage.

This risk assessment supports the licensee amendment request for a one-time five year extension of the ILRT interval. Callaway Plant is currently committed to 10 CFR 50 Appendix J Option B through a statement of compliance with Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, in Technical Specification 5.5.16. This Regulatory Guide was developed to endorse NEI 94-01, Revision 0, with certain modifications and additions.

In its Final Safety Evaluation dated June 25, 2008, the NRC has accepted, with specific limitations, topical report NEI 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and EPRI Report No. 1009325, Revision 2, August 2007, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," for referencing in plant Technical Specifications to permanently extend the ILRT surveillance interval to 15 years. Callaway Plant is not, at this time, applying for a permanent 15-year ILRT surveillance interval. However, in the NRC Final Safety Evaluation, dated June 25, 2008, Section 3.2, it is stated that the guidance provided in EPRI Report No. 1009325, Revision 2, for Probabilistic Risk Assessment (PRA) modeling is substantially the same as found in the NEI interim guidance/ methodology used to support one-time, 15-year ILRT extensions.

As a result, the risk assessment (provided in Attachment 4) which follows the guidelines from NEI 94-01, Revision 2, effectively follows the methodology used in EPRI Report No. 1009325, Revision 2, as well as the NRC regulatory guidance on the use of PRA findings and risk insights as outlined in RG 1.174.

The risk assessment results for Callaway Plant are consistent with those of previous studies supporting other plants' ILRT extension requests. The results below demonstrate that the increases in risk and large early release frequency (LERF) resulting from the proposed amendment are within established RG 1.174

guidelines and that defense-in-depth principles would be maintained. The complete Callaway Plant risk assessment is provided in Attachment 4.

The following are the conclusions from the completed risk assessment associated with a one-time five-year deferral of the Type A ILRT test interval from 10 years to 15 years:

4.5.1 LERF

RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-5}/yr and increases in large early release frequency (LERF) below 10^{-6}/yr. Since the ILRT does not impact CDF, the relevant risk metric in this application is LERF and the relevant criterion is the change in LERF. The increase in LERF resulting from a change in the Type A (ILRT) test interval from one in ten years (current interval) to one in fifteen years (proposed interval) is conservatively estimated as 1.66E-07/yr using the NEI guidance. The increase in LERF resulting from three in ten years (original interval) to one in fifteen years (proposed interval) is conservatively estimated as 3.98E-07/yr. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a small change to the Callaway Plant risk profile based on the RG 1.174 definition.

RG 1.174 also states that when the calculated increase in LERF is in the range of 1.0E-6 per reactor year to 1.0E-7 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-5 per reactor year. Therefore, an additional estimation of the impact from external events was made. In this case, the total LERF was conservatively estimated as 2.20E-06 per reactor year for Callaway. This is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-5.

4.5.2 Total Population Dose

The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.02 person-rem/yr when compared with the current Type A test frequency (once-per-ten-years), and 0.06 person-rem/yr when compared with the original Type A test frequency (three-in-ten-years). The NRC Final Safety Evaluation and EPRI Report No. 1009325, Revision 2-A, state that a small increase in population dose is defined as an increase of $\leq$ 1.0 person-rem per year or $\leq$ 1% of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The risk increase of this change is well below the above acceptance criterion and therefore is determined to be small.

4.5.3 CCFP

The increase in the conditional containment failure probability (CCFP) from the one test in ten years interval (current interval) to one test in fifteen years interval (proposed interval) is 0.39%. The increase from the three tests in ten years interval (original interval) to one test in fifteen years interval is 0.93%. The NRC Final Safety Evaluation and EPRI Report No. 1009325, Revision 2-A, state that increases in CCFP of $\leq$ 1.5 percentage point are small for the risk impact assessment of the extended ILRT intervals. Therefore this increase is judged to be small and reflects defense-in-depth.

4.5.4 Risk Implications of Undetected Corrosion-Induced Leakage of Steel Liner

The results show that including corrosion effects calculated using the assumptions described in Section 4.4 of Attachment 4 does not significantly affect the results of the ILRT extension risk assessment. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates, with very conservative assumptions for all of the key parameters, yield increases in LERF of only 1.48E-07/yr due to corrosion. The results indicate that, even with very conservative assumptions, the conclusions from the base analysis would not change.

4.5.5 Compliance of Callaway PRA to Regulatory Guide 1.200

To address this ILRT extension request, Callaway Plant has performed a self assessment of the Callaway PRA to evaluate the level of compliance with criteria contained in RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated January 2007. This self assessment is provided as Attachment 5 to this submittal. The ILRT application was determined to be a Capability Category I application of the RG 1.200, Revision 1, criteria. This is based on the requirement for numerical results for CDF and LERF to determine the risk impact of the requested change and the fact that this change is risk-informed, not risk-based. In the self assessment each of the current PRA supporting requirements (SRs) that did not comply with RG 1.200 Category II criteria is listed along with the assessment and evaluation of the non-conforming SR that shows that it has no material impact on the ILRT interval extension request.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

As described above, this amendment application involves a requested change to Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," to reflect a one-time five-year deferral of the containment Type A integrated leak rate test (ILRT) from once in ten years to once in 15 years.

AmerenUE has evaluated whether or not a significant hazards analysis is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," Section (c) as discussed below:

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

   Response: No

   The proposed change will revise Callaway Plant TS 5.5.16, "Containment Leakage Rate Testing Program," to reflect a one-time, five-year extension for the containment Type A test date to enable the implementation of a 15-year test interval. While the containment is designed to contain radioactive material that may be released from the reactor core following a design basis Loss-of-Coolant Accident (LOCA), the test interval associated with Type A testing is part of ensuring the plant's ability to mitigate the consequences of accidents described in the FSAR and does not involve a precursor or initiator of any accident previously evaluated. Thus, the proposed change to the Type A test interval cannot increase the probability of an accident previously evaluated in the FSAR.

   Type A testing does provide assurance that the containment will not exceed allowable leakage rate criteria specified in the TS and will continue to perform its design function following an accident. However, per NUREG-1493, "Performance-Based Containment Leak-Test Program," Type A tests identify only a few potential leakage paths that cannot be identified by Type B and C testing. The current Type B and C penetration test frequencies for Callaway are established based on performance, using the requirements of 10 CFR 50, Appendix J, Option B, and the Type B and C testing requirements will not be changed as a result of the proposed license amendment. As a result, with respect to the consequences of an accident, a risk assessment of the proposed change has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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

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

   Response: No

   The proposed change is for a one-time, five-year extension of the Type A test for Callaway Plant and will not affect the control parameters governing unit operation or the response of plant equipment to transient or accident conditions. The proposed change does not introduce new equipment, modes of system operation, or failure mechanisms.

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

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The Callaway Plant containment consists of the concrete containment building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis LOCA. Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

   The containment is a prestressed, reinforced concrete, cylindrical structure with a hemispherical dome and a reinforced concrete base slab. The inside structure is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. A post-tensioning system is used to prestress the cylindrical shell and dome.

   The concrete containment building is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage-limiting boundary of the containment. Maintaining operability of the containment will limit leakage of fission product radioactivity released from the containment to the environment.

   The integrity of the containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the containment is verified by an ILRT, as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

   The existing 10-year interval at Callaway Plant is based on past performance. Previous Type A tests conducted at Callaway Plant indicate that leakage from containment has been less than all 10 CFR 50 Appendix J, Option B, leakage limits.

The proposed change for a one-time extension of the Type A test does not affect the method for Type A, B, or C testing or the test acceptance criteria. Type B and C testing will continue to be performed at the frequency required by Callaway Plant Technical Specifications. The containment inspections that are performed in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection," and 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," provide a high degree of a assurance that the containment will not degrade in a manner that is only detectable by Type A testing.

In NUREG-1493, "Performance-Based Containment Leak-Test Program," the NRC indicated that a 20-year extension for Type A testing resulted in an imperceptible increase in risk to the public. The NUREG-1493 study also concluded that, generically, the design containment leak rate contributes a very small amount to the individual risk and that the decrease in Type A testing frequency would have a minimal affect on this risk. AmerenUE has conducted risk assessments to determine the impact of a one-time change to the Callaway Plant Type A test schedule from a baseline value of once in 10 years to once in 15 years for the risk measures of Large Early Release Frequency (LERF), Total Population Dose, and Conditional Containment Failure Probability (CCFP). The results of the risk assessments indicate that the proposed change to the Callaway Plant Type A test schedule has a minimal impact on public risk.

Based on the above and on previous Type A test results for the Callaway Plant containment, the current containment surveillance program, and the results of the AmerenUE risk assessment, there is no reduction in the effectiveness of the Callaway Plant containment as a barrier to the release of the post-accident containment atmosphere to the public or to personnel in the Control Room. Thus, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above evaluations, AmerenUE concludes that the activities associated with the above changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC of "no significant hazards consideration" is justified.

5.2 Regulatory Requirements & Guidance

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in a plant's Technical Specifications (TSs). 10 CFR 36(c)(5), "Administrative controls," requires provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner will be included in a plant's TSs.

Additionally, 10 CFR 50, Appendix J, Option B, Section V, "Application," specifies that the regulatory guide (i.e., RG 1.163) or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TS. Deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant's TS.

The proposed change will revise Callaway TS 5.5.16 to reflect a one-time extension of the program requirements for the Type A test interval. The one-time extension deviates from the guidelines contained in RG 1.163 and NEI 94-01. Thus, this license amendment request is being submitted pursuant to 10 CFR 50, Appendix J, Option B, Section V.B.

Additionally, in accordance with 10 CFR 50, Appendix J, Option B, Section V, the proposed change to Callaway Plant TS does not require a supporting request for an exemption to Option B of Appendix J, in accordance with 10 CFR 50.12, "Specific exemptions."

6.0 ENVIRONMENTAL CONSIDERATION

AmerenUE has evaluated this proposed license amendment consistent with the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." AmerenUE has determined that this proposed change meets the criteria for categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and has determined that no irreversible consequences exist in accordance with paragraph (b) of 10 CFR 50.92, "Issuance of amendment." This determination is based on the fact that this change is being processed as an amendment to the license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or which changes an inspection or surveillance requirement and the amendment meets the following specific criteria :

1. **The amendment involves no significant hazards consideration.**

   As demonstrated in Section 5.1, "No Significant Hazards Consideration," above, the proposed change does not involve any significant hazards consideration.

2. **There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.**

   The proposed change will revise Callaway TS 5.5.16, "Containment Leakage Rate Testing Program" to reflect a one-time, five-year extension of the containment Type A test interval. The proposed change does not result in an increase in power level, and does not increase the production nor alter the flow path or method of disposal of radioactive waste or byproducts; thus, there will be no change in the amounts of radiological effluents released offsite.

   Based on the above evaluation, the proposed change will not result in a significant change in the types or significant increase in the amounts of any effluent released offsite.

3. **There is no significant increase in individual or cumulative occupational radiation exposure.**

   The proposed change will revise Callaway TS 5.5.16, "Containment Leakage Rate Testing Program" to reflect a one-time, five-year extension of the containment Type A test interval. The proposed change will not cause a change in the level of controls or methodology used for the processing of radioactive effluents or handling of solid radioactive waste, nor will the proposed amendment result in any change in the normal radiation levels in the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

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.

7.0 PRECEDENT

The NRC has previously approved similar license amendments for the following nuclear plants on the noted date. The proposed Callaway License Amendment Request is consistent with the previous amendment requests.

Indian Point Nuclear Generating, Unit 3 (April 17, 2001)
Crystal River, Unit 3 (August 30, 2001)
Peach Bottom Atomic Power Station, Unit 3 (October 4, 2001)
Oconee Nuclear Station, Unit 3 (February 28, 2002)
Susquehanna Steam Electric Station, Units 1 and 2 (March 8, 2002)
Seabrook Station, Unit 1 (April 11, 2002)
Calvert Cliffs Nuclear Power Plant, Unit 1 (May 1, 2002)
Indian Point Nuclear Generating, Unit 2 (August 5, 2002)
Comanche Peak Steam Electric Station, Units 1 and 2 (August 15, 2002)
North Anna Power Station, Unit 1 (December 31, 2002)
Beaver Valley Power Station, Units 1 and 2 (March 5, 2003)
River Bend Station, Unit 1 (March 5, 2003)
McGuire Nuclear Station, Units 1 and 2 (March 12, 2003)
Catawba Nuclear Station, Units 1 and 2 (March 12, 2003)
Joseph M. Farley Nuclear Plant, Units 1 and 2 (March 21, 2003)
Duane Arnold Energy Center (March 21, 2003)
Monticello Nuclear Generating Plant (March 31, 2003)
Perry Nuclear Power Plant, Unit 1 (April 8, 2003)
Hope Creek Generating Station (April 16, 2003)
Sequoyah Nuclear Plant, Units 1 and 2 (May 29, 2003)
Three Mile Island Nuclear Station, Unit 1 (August 14, 2003)
Fort Calhoun Station, Unit 1 (August 15, 2003)
LaSalle County Station, Units 1 and 2 (November 19, 2003)
Clinton Power Station (January 8, 2004)
Vogtle Electric Generating Plant, Units 1 and 2 (January 12, 2004)
Grand Gulf Nuclear Station, Unit 1 (January 28, 2004)
H. B. Robinson Steam Electric Plant, Unit 2 (February 11, 2004)
Quad Cities Nuclear Power Station, Units 1 and 2 (March 8, 2004)
Kewaunee Nuclear Power Plant (April 6, 2004)
James A. Fitzpatrick Nuclear Power Plant (September 28, 2004)
Dresden Nuclear Power Station, Units 2 and 3 (October 13, 2004)
Edwin I. Hatch Nuclear Plant, Unit 2 (February 1, 2005)
Browns Ferry Nuclear Plant, Units 2 and 3 (March 9, 2005)
Pilgrim Nuclear Power Station (March 30, 2005)
Millstone Power Station, Unit 2 (April 6, 2005)
Columbia Generating Station (April 12, 2005)
Vermont Yankee Nuclear Power Station (August 31, 2005)
R. E. Ginna Nuclear Power Plant (December 8, 2005)
St. Lucie Plant, Unit 2 (December 23, 2005)
Shearon Harris Nuclear Power Plant, Unit 1 (March 30, 2006)

Watts Bar Nuclear Plant, Unit 1 (August 22, 2006)
Prairie Island Nuclear Generating Plant, Units 1 and 2 (October 2, 2006)
Cooper Nuclear Station (October 3, 2006)
Millstone Power Station, Unit 3 (June 29, 2007)
Byron Station, Units 1 and 2 (February 12, 2008)
Braidwood Station, Units 1 and 2 (April 2, 2008)
Limerick Nuclear Generating Station, Units 1 and 2 (February 20, 2008)
Point Beach Nuclear Plant, Units 1 and 2 (February 26, 2008)
Surry Power Station, Unit 2 (December 18, 2008)

8.0 REFERENCES

1. Letter from A. Pietrangelo (NEI) to NEI Administrative Points-of-Contact, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals," dated November 13, 2001.

2. Letter from A. Pietrangelo (NEI) to NEI Administrative Points-of-Contact, "One-Time Extension of Containment Integrated Leak Rate Test Interval - Additional Information," dated November 30, 2001.

3. Letter from C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to USNRC, "Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," dated March 27, 2002.

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ATTACHMENT 2

MARKED-UP TECHNICAL SPECIFICATION PAGE

5.5.15 Safety Function Determination Program (SFDP) (continued)

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

   1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

(continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

2. The visual examination of steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

3. The unit is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement during the Refuel 14 outage (fall of 2005).

INSERT 1

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.1 psig.

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

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests;

2. Air lock testing acceptance criteria are:

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

   b) For each door, leakage rate is $\leq 0.005\ L_a$ when pressurized to ≥ 10 psig.

e. The provisions or Technical Specifications SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of Technical Specification SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

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4. The first Type A test performed after the October 26, 1999 Type A test shall be performed no later than October 25, 2014.

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ATTACHMENT 3

RETYPED TECHNICAL SPECIFICATION PAGE

5.5.15 Safety Function Determination Program (SFDP) (continued)

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

(continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

2. The visual examination of steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

3. The unit is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement during the Refuel 14 outage (fall of 2005).

4. The first Type A test performed after the October 26, 1999 Type A test shall be performed no later than October 25, 2014.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.1 psig.

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

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests;

2. Air lock testing acceptance criteria are:

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

b) For each door, leakage rate is $\leq 0.005\ L_a$ when pressurized to $\geq$ 10 psig.

e. The provisions of Technical Specifications SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of Technical Specification SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

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ATTACHMENT 4

RISK ASSESSMENT FOR CALLAWAY PLANT TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

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RISK ASSESSMENT FOR CALLAWAY PLANT TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

1.0 PURPOSE OF ANALYSIS

The purpose of this analysis is to provide a risk assessment of a one-time extension of the currently allowed containment Type A integrated leak rate test (ILRT) surveillance interval from one test in ten years to one test in fifteen years. The extension of the ILRT interval would allow for substantial cost savings as the ILRT could be deferred to a later scheduled refueling outage for the Callaway Plant.

The risk assessment follows the guidelines from Nuclear Energy Institute (NEI) 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," [1], the methodology used in Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," [2], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide (RG) 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," as applied to ILRT interval extensions and risk insights in support of a request for a plant's licensing basis as outlined in RG 1.174, "An Approach for Using Probabilistic Rick Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Bases," [3]. The methodology used in EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," [4], the NEI guidance document, "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," dated November 2001 [5], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implication of corrosion-induced leakage of steel liners going undetected during the extended test interval [6] were also used in this risk assessment.

In 2008 the NRC issued a Final Safety Evaluation [7] which accepted, with specific limitations, NEI 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2. Although, Callaway Plant is not, at this time, applying for a permanent 15 year ILRT surveillance interval, in the NRC Final Safety Evaluation, Section 3.2, it is stated that the guidance provided in EPRI Report No. 1009325, Revision 2, for PRA modeling is substantially the same as found in the NEI interim guidance / methodology used to support one-time, 15-year ILRT extensions. Thus, it is appropriate to use the guidance / methodology provided in EPRI Report No. 1009325, Revision 2, for Callaway's ILRT one-time extension application.

1.1 Background

The revision to 10CFR50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," that incorporated Option B, "Performance-Based Requirements," allows individual plants to extend the ILRT (Type A) surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A frequency is based on acceptable performance history defined as two consecutive periodic Type A tests, at least 24 months apart, in which the calculated performance leakage rate was less than normal containment leakage rate of 1.0 L_a.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 [8], provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study.

The results of that study are documented in EPRI Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative Pressurized Water Reactor (PWR) plant (i.e., Surry) containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Accordingly, for Callaway Plant, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for the facility.

The guidance provided in Appendix H of EPRI Report No. 1009325, Revision 2 [2], for performing risk impact assessments in support of ILRT extensions builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, "Inservice Inspection." More specifically, Subsection IWE, "Requirements for Class NMC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulation 10 CFR 50.55a(b)(2)(ix)(E) requires licensees to conduct visual inspections of the accessible areas of the interior of the containment. Additionally, Type B and C local leak rate tests (LLRTs) performed to verify the leak-tight integrity of containment penetrations, airlocks, seals, gaskets, and containment isolation valves are not affected by the change to the Type A test frequency.

1.2 Criteria

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF (Callaway does not credit containment over-pressure for the mitigation of design basis accidents), the relevant risk metric in this application is LERF and the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year. RG1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure, and show, that key principles such as defense-in-depth philosophy are met. Therefore, additional risk metrics, specifically the increase in population dose and the increase in the conditional containment failure probability (CCFP), are also evaluated to help ensure that the key safety principles in RG 1.174 are met.

Regarding CCFP, changes of up to 1.1% have been accepted by the NRC for the one-time requests for extension of ILRT intervals. The NRC Final Safety Evaluation [7] states that a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. The Final Safety Evaluation defines increases in CCFP $\leq$1.5 percentage point as small changes for the risk impact assessment of the proposed extended ILRT interval.

The population dose (person-rem per year) is examined to demonstrate the relative change in this parameter. While no acceptance guidelines for this additional figure of merit are published, examinations of NUREG-1493 and the Final Safety Evaluation indicate a range of incremental increase in population dose that have been accepted by the NRC. The range of incremental population dose increase is from ≤0.01 to 0.2 person-rem/yr and/or 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (NUREG-1493 [8], Figure 7-2) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a small increase in population dose is defined as an increase in population dose of ≤1.0 person-rem per year or 1% of the total population dose, whichever is less restrictive, for the risk impact assessment of the proposed extended ILRT interval.

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in Appendix H of EPRI Report No. 1009325, Revision 2 [2], EPRI TR-104285 [4], NUREG-1493 [8], and the Calvert Cliffs containment liner corrosion analysis [6]. The analysis uses the current Callaway Plant PRA model (individual plant examination (IPE) Revision 5) [9] that includes a Level 2 analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no or negligible release). This risk assessment is applicable to Callaway.

There are six general steps included in this assessment, as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.
2. Develop plant-specific population dose (person-rem per reactor year) for each of the eight containment release scenario types from the Callaway specific consequence analyses.
3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.
4. Determine the change in risk in terms of LERF in accordance with RG 1.174 [3] and compare with the acceptance guidelines of RG 1.174.
5. Determine the impact on the CCFP.
6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, to external events, and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore, consistent with the other industry containment leak risk assessments, the Callaway assessment evaluates LERF and delta LERF in accordance with the risk acceptance guidance of RG 1.174. Changes in population dose and conditional containment failure probability are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved. Callaway does not credit containment over-pressure in the ECCS recirculation analysis. As CDF is not impacted by the Type A test, the relevant risk metric in this application is LERF and the relevant LERF criterion is used. This evaluation for Callaway uses ground rules and methods to calculate changes in risk metrics that are similar to those used in Appendix H of EPRI Report No. 1009325, Revision 2 [2].

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The technical adequacy of the Callaway PRA is consistent with the requirement of RG 1.200 as is relevant to this ILRT interval extension application.
- The Callaway Level 1 and Level 2 internal events PRA models provide representative results.
- It is appropriate to use the Callaway internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increase in population dose) will not substantially differ if external events like fire, seismic and internal flooding were to be included in the calculations.
- Dose results for the containment failures modeled in the PRA can be characterized by information provided in NUREG/CR-4551 [10]. They are estimated by scaling the NUREG/CR-4551 results by population differences for Callaway compared to the NUREG/CR-4551 reference plant.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [4] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is 1 L_a. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is 10 L_a, based on the previously approved methodology performed for Indian Point Unit 3 [11, 12] and the NRC Final Safety Evaluation [7].
- The representative containment leakage for Class 3b sequences is 100 L_a, based on the guidance provided in EPRI Report No. 1009325, Revision 2-A [30], and the NRC Final Safety Evaluation [7].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [11, 12]. The increase in the LERF from the increase in the length of the ILRT test interval can be represented by the increase in the Class 3b frequency based on the NRC Final Safety Evaluation [7].
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- Callaway currently has no quantitative fire or seismic PRA analyses. The methods used in this risk assessment to obtain the impact from internal events can not be applied for the external events. A suitable estimate of the risk impact from the external events is performed by assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

4.0 INPUTS

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 General Resources Available

Various industry studies on containment leakage risk assessment are briefly summarized here:

1. NUREG/CR-3539 [13]
2. NUREG/CR-4220 [14]

3. NUREG-1273 [15]
4. NUREG/CR-4330 [16]
5. EPRI TR-105189 [17]
6. NUREG-1493 [8]
7. EPRI TR-104285 [4]
8. NUREG-1150 [18] and NUREG/CR-4551 [10]
9. NEI Interim Guidance [5][20]
10. Calvert Cliffs liner corrosion analysis [6]
11. EPRI Report No. 1009325, Revision 2, Appendix H

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to the second study, NUREG/CR-4220, that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the basis for the consequence analysis of the ILRT interval extension for Callaway. The ninth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the eleventh study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent 15-year extension of the ILRT interval.

NUREG/CR-3539 [13]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [19] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on light water reactor (LWR) accident risks is relatively small.

NUREG/CR-4220 [14]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand Licensee Event Reports (LERs), ILRT reports and other related records to calculate the unavailability of containment due to leakage.

NUREG-1273 [15]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [16]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189 [17]

The EPRI TR-105189 study is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains a quantitative evaluation (using the EPRI ORAM software) of the impact of extending ILRT and LLRT intervals on shutdown risk for two reference plants (a Boiling Water Reactor (BWR)-4 and a PWR). The conclusion from the study is that a small but measurable safety benefit is realized from extending the test intervals.

NUREG-1493 [8]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

> Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.

Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [4]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models for the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failure dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures

6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

> "... the proposed CLRT [containment leak rate tests] frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.04 person-rem per year ..."

The detailed definitions of the eight accident classes in this report are provided in Table 4.1.1. These containment failure classifications were used in this analysis to determine the risk impact of extending the Containment Type A test interval, as described in Section 5 of this report.

NUREG -1150 [18] and NUREG/CR-4551 [10]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. The collapsed APBs defined in NUREG/CR-4551 are characterized by five attributes related to the accident progression. Unique combinations of the five attributes result in a set of seven bins that are relevant to the analysis. Table 4.1.2 provides a description of the collapsed APBs from NUREG/CR-4551.

With the Callaway Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent Callaway. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

Table 4.1.1
EPRI/NEI Containment Failure Classifications

Accident Class	Description
1	**Containment remains Intact.** Include accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values La, under Appendix J for that plant.
2	**Containment isolation failures.** Include those accidents in which there is a failure to isolate the containment.
3	**Independent (or random) isolation failures (Type A test).** Include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.

Table 4.1.1 EPRI/NEI Containment Failure Classifications	
4	**Independent (or random) isolation failures (Type B test).** Include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	**Independent (or random) isolation failures (Type C test).** Include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	**Containment isolation failures (dependent failures personnel errors)** Include those leak paths covered in the plant test and Maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	**Severe accident phenomena induced failures (early and late containment failures).** Include the accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	**Containment bypass (SGTR, MSIV leakage and ISLOCA)** Include those accidents in which the containment is bypassed (either as an initial condition or induced by phenomena). Changes in Appendix J testing requirements do not impact these accidents.

Table 4.1.2 Summary Accident Progression Bin (APB) Descriptions (NUREG/CR-4551, Surry)	
APB Number	**Description**
1	**CD, VB, Early CF, Alpha Mode** Core damage (CD) occurs followed by a very energetic molten fuel-coolant interaction in the vessel; the vessel fails and generates a missile that fails the containment as well. Includes accidents that have an Alpha mode failure of the vessel and the containment except those follow Event V or an SGTR. It includes Alpha mode failures that follow isolation failures because the Alpha mode containment failure is of rupture size.
2	**CD, VB, Early CF, RCS Pressure > 200 psia** Core damage occurs followed by vessel breach (VB). Implies early containment failure (Early CF) with the RCS above 200 psia when the vessel fails. Early CF means at or before VB, so it includes isolation failures and seismic containment failures at the start of the accident as well as containment failure at VB. It does not include bins in which containment failure at VB follows Event V or an SGTR, or Alpha mode failures.

Table 4.1.2 Summary Accident Progression Bin (APB) Descriptions (NUREG/CR-4551, Surry)	
3	**CD, VB, Early CF, RCS Pressure < 200 psia** Core damage occurs followed by vessel breach. Implies Early CF with the RCS below 200 psia when the vessel fails. It does not include bins in which the cone containment failure at VB or an SGTR or Alpha mode failures.
4	**CD, VB, Late CF** Core damage occurs followed by vessel breach. Includes accidents in which the containment was not failed or bypassed before the onset of core-concrete interaction (CCI) and in which the vessel failed. The failure mechanisms are hydrogen combustion during CCI, Basemat Melt-Through (BMT) in several days, or eventual overpressure due to the failure to provide containment heat removal in the days following the accident.
5	**CD, Bypass** Core damage occurs followed by vessel breach. Includes Event V and SGTRs no matter what happens to the containment after the start of the accident. It also includes SGTRs that do not result in VB.
6	**CD, VB, No CF** Core damage occurs followed by vessel breach. Includes accidents not evaluated in one of the previous bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.
7	**CD, No VB** Core damage occurs but is arrested in time to prevent vessel breach. Includes accident progressions that avoid vessel failures except those that bypass the containment. Most of the bins placed in this bin have no containment failure as well as no VB. It also includes bins in which the containment is not isolated at the start of the accident and the core is brought to a safe stable state before the vessel fails.

<u>NEI Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [5][20]</u>

The guidance provided in this document builds on the EPRI risk impact assessment methodology [4] and the NRC performance-based containment leakage test program [8], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

<u>Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [6]</u>

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms was factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. Since Callaway has a similar type of containment, the Calvert Cliffs methodology was used to perform the liner corrosion analysis in this risk assessment.

EPRI Report No. 1009325, Revision 2, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [2]

This report provides a generally applicable assessment of the risk involved in extension of ILRT test intervals to permanent 15-year intervals. Appendix H of this document provides guidance for performing plant-specific supplemental risk impact assessments and builds on the previous EPRI risk impact assessment methodology [4] and the NRC performance-based containment leakage test program [8], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC Safety Evaluation Report (SER)) and Crystal River.

The approach included in this guidance document was used in the Callaway assessment to determine the estimated increase in risk associated with the ILRT extension. This document includes the base for the values assigned in determining the probability of leakage for the EPRI Class 3a and 3b scenarios in this analysis as described in Section 5.

4.2 Plant-Specific Inputs

The plant-specific information that was used to perform the Callaway ILRT Extension Risk Assessment includes the following:

- Callaway PRA Model [25]
- Release category definitions used in the Level 2 Model [25]
- Population within a 50-mile radius of Callaway [26]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [27, 28]. The current Type A test interval is 10 years.

Callaway PRA Model

The original Callaway PRA model was developed to satisfy NRC's Generic Letter 88-20 requirement that each licensee perform an "individual plant examination" (IPE) to search for plant-specific severe accident vulnerabilities. Results of the Callaway PRA were submitted to the NRC, pursuant to this requirement, in September of 1992. The NRC SER on the Callaway IPE submittal was issued in May 1996. Since completion of the Callaway IPE (PRA), the model has been periodically updated to incorporate plant design changes, model changes, new failure and test/maintenance data, and industry information so that the model is technically adequate and is characteristic of the as-built plant. The latest update, Fourth PRA Update or IPE Revision 5 [25], was completed in 2006. The total internal events CDF excluding internal flooding was quantified as 4.27E-5 /yr in Callaway PRA Evaluation PRAER 08-322 [29].

Association of Callaway specific source term categories with EPRI accident classes and NUREG/CR-4551 collapsed accident progression bins

In the EPRI methodology to determine the risk impact of extending the Containment Type A test interval, the EPRI accident classes in Table 4.1.1 are used to define the spectrum of plant releases, and the NUREG/CR-4551 collapsed APBs in Table 4.1.2 are used to define the accident progression for the population dose calculation. A major factor related to the use of the EPRI methodology and NUREG/CR-4551 in this evaluation is that the results of the Callaway PRA model are not defined in the same terms as reported in NUREG/CR-4551 and the EPRI report. In order to use the EPRI methodology and the Surry Level 3 model and results, Callaway PRA Evaluation PRAER 08-322 [29] was performed to match the

Callaway PRA source term categories (STCs) from Callaway Level 2 PRA analysis to the EPRI accident classes and the collapsed APBs. Table 4.2.1 presents the results of association of Callaway STCs to EPRI accident classes and NUREG/CR-4551 APBs from PRAER 08-322. The twenty-five source term categories (STCs) are summarized in Table 4.2.2 for reference purposes.

Table 4.2.1
Association of Callaway Source Term Categories to EPRI Accident Classes and NUREG/CR-4551 Accident Progression Bins

Callaway STC	Callaway Description	EPRI Accident Class	EPRI Description	NUREG/CR-4551 APB	NUREG Description
25	No release within 48 hour mission time	1	Containment intact	6, 7	No containment failure
N/A (1)	N/A	2	Large containment isolation failures	2	Early CF, RCS > 200 psia
3, 4, 5, 6, 7, 8, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24	Early and late containment failures (Alpha-mode, DCH, CHR failure, basemat melt through, etc.)	7	Severe accident phenomena-induced failures (early and late containment failures)	1, 3, 4	Alpha Mode, Early and Late CF
1, 2, 9, 10, 11	ISLOCA and SGTR	8	Containment bypass (SGTR, MSIV Leakage, and ISLOCA)	5	CD, Bypass

(1) The Callaway Level 2 PRA did not produce an STC for containment isolation failure due to the relative independence of the containment isolation system in Callaway IPE analysis [25]. Callaway PRA Evaluation PRAER 08-322 [29] estimated the current release frequency associated with containment isolation failure.

Table 4.2.2
Callaway Level 2 PRA Model Source Term Category (STC)

STC	Definition
1	ISLOCA, Release Reduction Factors NOT Effective
2	ISLOCA, Release Reduction Factors Effective
3	Alpha Mode Failure, Containment Spray NOT Effective
4	Alpha Mode Failure, Containment Spray Effective

Table 4.2.2 Callaway Level 2 PRA Model Source Term Category (STC)	
5	Early Rupture, Containment Spray NOT Effective, Release Reduction Factors NOT Effective
6	Early Rupture, Containment Spray NOT Effective, Release Reduction Factors Effective
7	Early Rupture, Containment Spray Effective, Release Reduction Factors NOT Effective
8	Early Rupture, Containment Spray Effective, Release Reduction Factors Effective
9	SGTR, Release Reduction Factors NOT Effective
10	SGTR, Release Reduction Factors Moderate Effective
11	SGTR, Release Reduction Factors Effective
12	Early Leak, Containment Spray NOT Effective, Release Reduction Factors NOT Effective
13	Early Leak, Containment Spray NOT Effective, Release Reduction Factors Effective
14	Early Leak, Containment Spray Effective, Release Reduction Factors NOT Effective
15	Early Leak, Containment Spray Effective, Release Reduction Factors Effective
16	Late Rupture, Containment Spray NOT Effective, Release Reduction Factors NOT Effective
17	Late Rupture, Containment Spray NOT Effective, Release Reduction Factors Effective
18	Late Rupture, Containment Spray Effective, Release Reduction Factors NOT Effective
19	Late Rupture, Containment Spray Effective, Release Reduction Factors Effective
20	Late Leak, Containment Spray NOT Effective, Release Reduction Factors NOT Effective
21	Late Leak, Containment Spray NOT Effective, Release Reduction Factors Effective
22	Late Leak, Containment Spray Effective, Release Reduction Factors NOT Effective
23	Late Leak, Containment Spray Effective, Release Reduction Factors Effective
24	Basemat Melt-through
25	No Release

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 [10] and adjusting the results for Callaway. From the data provided in NUREG/CR-4551, the population dose for the Surry Plant associated with each APB is calculated. Table 4.2.3, below, summarizes the calculation. The second column of the table, Mean Fractional Contribution to Risk (MFCR), is calculated from the average of two samples delineated in Table 5.1-3 of NUREG/CR-4551. The total population dose risk at 50 miles from internal events in person-rem/yr is provided as the average of two samples in Table 5.1.1 of NUREG/CR-4551. The population dose risk for each APB is the product of the total population dose risk and the mean fractional APB contribution. NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-3 [10]. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency. Finally the population dose at 50 miles for each APB is calculated by dividing the population dose risk shown in the third column of Table 4.2.3 by the collapsed bin frequency shown in the fourth column of Table 4.2.3.

Table 4.2.3
Calculation of Surry Plant Population Dose Risk within 50 Miles [10]

APB Number	Mean Fractional Contribution to Risk (MFCR)	Population Dose Risk (person-rem/yr, mean)	APB Frequency (per year)	Population Dose (person-rem)
1	0.029	0.158	1.23E-07	1.28E+06
2	0.019	0.106	1.64E-07	6.46E+05
3	0.002	0.013	2.01E-08	6.46E+05
4	0.216	1.199	2.42E-06	4.95E+05
5	0.732	4.060	5.00E-06	8.12E+05
6	0.001	0.006	1.42E-05	4.23E+02
7	0.002	0.011	1.91E-05	5.76E+02
Totals	1.000	5.55	4.10E-05	

Each of the Callaway source term categories was associated with an applicable Collapsed Accident Progression Bin (APB) from NUREG/CR-4551 as provided in Table 4.2.1. The results of population dose within 50-mile radius of Surry Plant in Table 4.2.3 can be used as an approximation of the equivalent Callaway population dose if it is corrected for allowable containment leak rate (L_a), reactor power level and the population density surrounding Callaway.

The L_a adjustment is applicable only to those APBs affected by normal leakage. L_a for Callaway is 0.2%w/o/day [27] and L_a for Surry is 0.1%w/o/day.

$F_{leakage}$ = *L_a of plant (%w/o/day) / L_a of reference plant*
= 0.2%w/o/day / 0.1%w/o/day
= 2

The rated power level for Callaway is 3565 MW_t [26] and the rated power level for Surry is 2441 MW_t.

$F_{PowerLevel}$ = *Rated power level of plant (MW_t) / Rated power level of reference plant*
= 3565 MW_t / 2441 MW_t
= 1.46

The total population within a 50-mile radius of Callaway is 4.91E+05 [26]. This population value is compared to the population value that is provided in NUREG/CR-4551 (1.23E+06, Surry) in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for Callaway.

$F_{Population}$ = *Population50Miles of plant / Population50Miles of reference plant*
= 4.91E+05 / 1.23E+06
= 0.40

The factors developed above are used to adjust the population dose for the Surry plant for Callaway. For intact containment end states, the total population dose factor is as follows:

$$F_{Intact} = F_{leakage} * F_{PowerLevel} * F_{Population}$$
$$= 2 * 1.46 * 0.40$$
$$= 1.17$$

For EPRI accident classes not dependent on containment leakage, the population dose factor is as follows:

$$F_{others} = F_{PowerLevel} * F_{Population}$$
$$= 1.46 * 0.40$$
$$= 0.58$$

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. The above adjustments provide an approximation for Callaway of the population doses associated with each of the release categories from NUREG/CR-4551.

Table 4.2.4 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for Callaway.

Table 4.2.4
Calculation of Callaway Population Dose Risk within 50 Miles

APB Number	Surry Population Dose (person-rem)	Population Dose Factor	Callaway Population Dose (person-rem)
1	1.28E+06	0.58	7.49E+05
2	6.46E+05	0.58	3.77E+05
3	6.46E+05	0.58	3.77E+05
4	4.95E+05	0.58	2.89E+05
5	8.12E+05	1.17	9.47E+05
6	4.23E+02	1.17	4.93E+02
7	5.76E+02	1.17	6.72E+02

4.3 Impact of Extension on Detection of Component Failures That Lead to Leakage (Small and Large)

The ILRT can detect a number of component failures such as liner breach, failure of certain bellows arrangements, and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class, as defined in Table 4.1.1, is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined in accordance with the EPRI Guidance. For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., 2 "small" failures in 217 tests leads to 2/217 = 0.0092).

For Class 3b, Jeffery non-informative prior distribution is assumed for no "large" failures in 217 test (i.e., 0.5/(217+1) = 0.0023).

In this evaluation, Class 3b frequency is conservatively determined by multiplying the total CDF by its failure probability, 0.0023. Class 3a frequency is determined by multiplying the total CDF by the failure probability of Class 3a, 0.0092.

Consistent with the NEI Guidance [5], the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yr / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yr / 2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0 / 1.5) higher for the probability of a leak that is detectable only by ILRT testing. Accordingly, an extension of the ILRT interval to fifteen years can be estimated to lead to a factor of 5.0 (average time that a leak could exist without detection for fifteen-year interval is 7.5 years, and 7.5 / 1.5 = 5) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the Indian Point Unit 3 request for a one-time ILRT extension that was approved by the NRC [12]) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B and C local leak rate tests that will still occur). Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

4.4 Impact of Extension on Detection of Steel Liner Corrosion That Leads to Leakage

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [6]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. Callaway has a similar type of containment as described in the FSAR [26], Section 3.8.1.1.1:

> "The reactor building consists of a prestressed, reinforced concrete, cylindrical structure with a hemispherical dome and a conventionally reinforced concrete base slab ..." and "The base slab, cylinder, and dome are reinforced by bonded reinforcing steel ..."

The following approach is used to determine the change in likelihood, due to extending the ILRT interval, of detecting corrosion of the containment steel liner. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the containment cylinder and dome
- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

Table 4.4.1 summarizes the steel liner corrosion analysis (base case) for Callaway with the following assumptions:

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 4.4.1, Step 1.)
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to this Callaway containment analysis. These events, one at North

Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.

- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis), and there is no evidence that additional corrosion issues were identified. (See Table 4.4.1, Step 1.)
- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages. (See Table 4.4.1, Step 2 and 3) Sensitivity studies are included that address doubling this rate every ten years and every two years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. Conservative probabilities of 1% for the cylinder and dome and 0.1% for the basemat are used in this analysis, and sensitivity studies are included that increase and decrease the probabilities by an order of magnitude. (See Table 4.4.1, Step 4)
- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the containment cylinder and dome region. (See Table 4.4.1, Step 4)
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection. (See Table 4.4.1, Step 5) Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.
- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. Thus the probability of all non-detectible failures from the corrosion sensitivity analysis are added to the EPRI Class 3b (and subtracted from EPRI Class 1). This approach is conservative and avoids detailed analysis of containment failure timing and operator recovery actions.

Table 4.4.1
Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome	Containment Basemat
1	**Historical Steel Liner Flaw Likelihood** Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).	**2 Events** **2/(70*5.5) = 5.2E-3**	**0 Events (assume half a failure)** **0.5/(70*5.5) = 1.3E-3**

Table 4.4.1
Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome	Containment Basemat
[redacted]			
2	**Age Adjusted Steel Liner Flaw Likelihood** During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).	**Year — Failure Rate** **1 — 2.1E-3** **avg 5-10 — 5.2E-3** **15 — 1.4E-2** **15 year average = 6.27E-3**	**Year — Failure Rate** **1 — 5.0E-4** **avg 5-10 — 1.3E-3** **15 — 3.5E-3** **15 year average = 1.57E-3**
[redacted]			
3	**Flaw Likelihood at 3, 10, and 15 years** Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis -- See Table 6 of Reference [6]).	**0.71% (1 to 3 years)** **4.06% (1 to 10 years)** **9.40% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the desired presentation of the results.)	**0.18% (1 to 3 years)** **1.02% (1 to 10 years)** **2.35% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the desired presentation of the results.)
[redacted]			
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1% (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.10%**

Table 4.4.1 Steel Liner Corrosion Base Case			
Step	**Description**	**Containment Cylinder and Dome**	**Containment Basemat**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	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.	100% Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Step 3 * 4 * 5)	**0.71%*1%*10% = 7.1E-6 (at 3 years)** **4.1%*1%*10% = 4.1E-5 (at 10 years)** **9.4%*1%*10% = 9.4E-5 (at 15 years)**	**0.18%*0.1%*100% = 1.8E-6 (at 3 years)** **1.0%*0.1%*100% = 1.0E-5 (at 10 years)** **2.4%*0.1%*100% = 2.4E-5 (at 15 years)**
7	**Total Likelihood of Non-Detected Containment Leakage** (Cylinder and Dome + Basemat)	**7.1E-6 + 1.8E-6 = 8.9E-6 (at 3 years)** **4.1E-5 + 1.0E-5 = 5.1E-5 (at 10 years)** **9.4E-5 + 2.4E-5 = 1.2E-4 (at 15 years)**	

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder and dome and the containment basemat for Callaway, which is shown in Step 7 of the table.

The above factors are applied to those core damage accidents that have no associated LERF. For example, the 3-in-10 year case is calculated as follows:

- Per Table 5.2, the EPRI Class 3b frequency is 9.82E-08/yr.
- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as
  (4.27E-5/yr) * 8.9E-6 = 3.80E-10/yr
  where 8.9E-6 was previously shown, above, to be the cumulative likelihood of non-detected containment leakage due to corrosion at 3 years.
- The 3-in-10 year Class 3b frequency including the corrosion-induced concealed flaw issue is calculated as
  (9.82E-08/yr) + (3.80E-10/yr) = 9.86E-08/yr
- The 3-in-10 year Class 1 frequency is deducted from the original value by the corrosion-induced concealed flaw frequency.
  (1.90E-05/yr) - (3.80E-10/yr) = 1.90E-05/yr

5.0 RESULTS

The application of the approach based on the guidance contained in EPRI Report No. 1009325, revision 2, Appendix H [2], EPRI-TR-104285 [4] and previous risk assessment submittals on this subject [6, 11, 21, 22, 23] led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5.1 lists these accident classes and their simplified descriptions (see Table 4.1.1 for the accident class details).

The analysis performed examined Callaway-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the breakdown of the severe accidents contributing to risk was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage (EPRI TR-104285 Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. For example, a valve failing to close following a valve stroke test (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and failure induced by phenomena, early and late, (EPRI TR-104285 Class 7 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.

Table 5.1
Accident Classes

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (failure to seal - Type B)
5	Small Isolation Failures (failure to seal - Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End States (including very low and no release)

- Accident sequences involving small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are impacted by changes in Type B and C test intervals. These sequences are not impacted by the changes in the Type A test interval and are not specifically examined in this analysis.

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5.1.
Step 2 Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.
Step 3 Evaluate risk impact of extending Type A test interval from 3 to 15 years and 10 to 15 years.
Step 4 Determine the change in risk in terms of LERF in accordance with RG 1.174.
Step 5 Determine the impact on the (CCFP).

5.1 Step 1 - Quantify the Base-Line Risk in Terms of Frequency Per Reactor Year

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285). The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 5.1 were developed for Callaway by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 4.2.1, determining the frequencies for Class 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4.

The total Callaway internal events CDF, excluding flooding, is determined from the Fourth Update of Callaway PRA:

CDF_{total} = total internal events CDF (excluding flooding)
= Non-Floods CDF + RV rupture CDF + CDFISLOCA
= 4.22E-05/yr + 3.0E-07/yr + 1.73E-07/yr
= 4.27E-05/yr

<u>Class 1 Sequences</u>

This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The Class 1 frequency is initially determined from the Callaway PRA model. Since the EPRI/NEI Class 3a and 3b frequencies are not calculated from the PRA model but are considered to apply only to Class 1 type sequences, the final Class 1 frequency is obtained by subtracting the Class 3a and 3b frequencies, calculated below, from the initial Class 1 frequency.

$$F_{class\ 1} = CDF_{intact} - F_{class\ 3a} - F_{class\ 3b}$$

Class 2 Sequences

This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. As noted in Table 4.2.1, the Callaway Level 2 PRA did not produce a source term category (STC) for containment isolation failure. Instead, Callaway PRA Evaluation PRAER 08-322 [29] was performed to estimate the current release frequency associated with containment isolation failure for Callaway. The frequency for Class 2 is determined as:

$F_{class\ 2} = 7.54E\text{-}09/yr$

Class 3 Sequences

This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small (>1 L_a to 35 L_a) or large (> 35 L_a).

The failure probabilities of Class 3a and 3b are determined as follows:

$PROB_{class\ 3a}$ = probability of small pre-existing containment liner leakage
= 0.0092 (see Section 4.3)

$PROB_{class\ 3b}$ = probability of large pre-existing containment liner leakage
= 0.0023 (see Section 4.3)

The frequencies for Class 3a and 3b can be conservatively determined as:

$$F_{class\ 3a} = PROB_{class\ 3a} * CDF_{total} = 0.0092 * (4.27E\text{-}05/yr) = 3.93E\text{-}07/yr$$

$$F_{class\ 3b} = PROB_{class\ 3b} * CDF_{total} = 0.0023 * (4.27E\text{-}05/yr) = 9.82E\text{-}08/yr$$

For this analysis, the associated containment leakage for Class 3a is 10 L_a and for Class 3b is 100 L_a. These assignments are consistent with the NRC Final Safety Evaluation [7] and the guidance provided in EPRI Report No. 1009325, Revision 2-A [30].

Class 4 Sequences

This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences

This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type C test components occurs. Because these failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 6 Sequences

This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences

This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). The frequency for Class 7 is determined in Callaway PRA Evaluation PRAER 08-322 [29] as:

$$F_{class\ 7} = 2.26E\text{-}05/yr$$

Class 8 Sequences

This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency can thus be determined as:

$$\begin{aligned} F_{class\ 8} &= CDF_{ISLOCA} + CDF_{SGTR} \\ &= (1.73E\text{-}07/yr) + (4.21E\text{-}07/yr) \\ &= 5.94E\text{-}07/yr \end{aligned}$$

CDF_{intact} can be determined by subtracting the frequencies of EPRI classes 7 and 8 from the total CDF.

$$\begin{aligned} CDF_{intact} &= CDF_{total} - F_{class\ 7} - F_{class\ 8} \\ &= (4.27E\text{-}05/yr) - (2.26E\text{-}05/yr) - (5.94E\text{-}07/yr) \\ &= 1.95E\text{-}05/yr \end{aligned}$$

The frequency for accident Class 1 can then be determined:

$$\begin{aligned} F_{class\ 1} &= CDF_{intact} - F_{class\ 3a} - F_{class\ 3b} \\ &= (1.95E\text{-}05/yr) - (3.93E\text{-}07/yr) - (9.82E\text{-}08/yr) \\ &= 1.90E\text{-}05/yr \end{aligned}$$

Summary of Accident Class Sequences

In summary, the accident sequence frequencies that can lead to radionuclide release to the public were derived consistent with the definitions of accident classes defined in EPRI TR-104285, the NEI Interim Guidance, and EPRI Report No. 1009325. Table 5.2 summarizes these accident frequencies by accident class for Callaway. The results for the corrosion sensitivity case as discussed in Section 4.4 are presented under the "With Corrosion" column.

Table 5.2 Callaway Release Frequencies as a Function of Accident Class (Base Case)			
Accident Classes (Containment Release Type)	**Description**	**Frequency (per Rx-yr)**	
		Without Corrosion	**With Corrosion**
1	No Containment Failure	1.90E-05	1.90E-05
2	Large Isolation Failures (Failure to Close)	7.54E-09	7.54E-09
3a	Small Isolation Failures (liner breach)	3.93E-07	3.93E-07
3b	Large Isolation Failures (liner breach)	9.82E-08	9.86E-08
4	Small Isolation Failures (failure to seal - Type B)	N/A	N/A
5	Small Isolation Failures (failure to seal - Type C)	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	2.26E-05	2.26E-05
8	Bypass (Interfacing System LOCA)	5.94E-07	5.94E-07
CDF	All CET End States	4.27E-05	4.27E-05

5.2 Step 2 - Develop Plant-Specific Person-Rem Dose (Population Dose) Per Reactor Year

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences compared to the reference plant as described in Section 4.2. Table 4.2.4 summarizes the results of the adjusted population dose at 50 miles for Callaway as a function of the APB. With the association of the APBs and the EPRI accident classes, the releases can be applied to the EPRI/NEI containment failure classification as follows:

Class 1 Dose

Class 1 is assigned the dose from the "no containment failure" APBs from NUREG/CR-4551, i.e., APB #6 and APB #7. Class 1 is considered to cover only containment leakage for the assumed intact containment at the leakage limit of 1 L_a.

$$D_{class\ 1} = D_{APB\ 6} + D_{APB\ 7}$$
$$= 1.16E{+}03 \text{ person-rem}$$
$$= 1.0 \text{ La}$$

Class 2 Dose

Consistent with the guidance provided in EPRI Report, the Class 2, containment isolation failures, dose is assigned from APB #2 (Early CF).

$$D_{class\ 2} = D_{APB\ 2}$$
$$= 3.77E{+}05 \text{ person-rem}$$

Class 3 Dose

The Class 3a and 3b doses are $10*L_a$ and $100*L_a$, respectively. This is consistent with the guidance provided in EPRI Report No. 1009325, Revision 2-A and the NRC Final Safety Evaluation.

$$D_{class\ 3a} = 10 * L_a$$
$$= 10 * (1.16E+03 \text{ person-rem})$$
$$= 1.16E+04 \text{ person-rem}$$

$$D_{class\ 3b} = 100 * L_a$$
$$= 100 * (1.16E+03 \text{ person-rem})$$
$$= 1.16E+05 \text{ person-rem}$$

Class 7 Dose

The Class 7 dose is assigned from APB #1 (Alpha Mode), #3 and #4 (Early and Late CF).

$$D_{class\ 1} = D_{APB\ 1} + D_{APB\ 3} + D_{APB\ 4}$$
$$= (7.49E+05 \text{ person-rem}) + (3.77E+05 \text{ person-rem}) + (2.89E+05 \text{ person-rem})$$
$$= 1.41E+06 \text{ person-rem}$$

Class 8 Dose

Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from APB #5 (Bypass).

$$D_{class\ 8} = D_{APB\ 8}$$
$$= 9.47E+05 \text{ person-rem}$$

The population dose estimates derived for use in the risk evaluation for all EPRI accident classes are summarized in Table 5.3.

Table 5.3
Callaway Population Dose Estimates for Area Within 50 Miles

Accident Classes (Containment Release Type)	Description	Population Dose (person-rem)
1	No Containment Failure	1.16E+03
2	Large Isolation Failures (Failure to Close)	3.77E+05
3a	Small Isolation Failures (liner breach)	1.16E+04
3b	Large Isolation Failures (liner breach)	1.16E+05
4	Small Isolation Failures (failure to seal - Type B)	N/A
5	Small Isolation Failures (failure to seal - Type C)	N/A

Table 5.3 Callaway Population Dose Estimates for Area Within 50 Miles		
Accident Classes (Containment Release Type)	**Description**	**Population Dose (person-rem)**
6	Other Isolation Failures (e.g., dependent failures)	N/A
7	Failures Induced by Phenomena (Early and Late)	1.41E+06
8	Bypass (Interfacing System LOCA)	9.47E+05

The above dose estimates, when combined with the results presented in Table 5.2, yield the Callaway baseline mean consequence measures for each accident class. Table 5.4 provides a summary of the base case as well as the corrosion sensitivity case.

Table 5.4
Callaway Annual Dose as a Function of Accident Class
Characteristic of Conditions for ILRT Required 3/10 Years (Base Case)

Accident Classes (Containment Release Type)	Description	Population Dose (person-rem)	Without Corrosion		With Corrosion (1)	
			Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)	Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)
1	No Containment Failure (2)	1.16E+03	1.90E-05	2.21E-02	1.90E-05	2.21E-02
2	Large Isolation Failures (Failure to Close)	3.77E+05	7.54E-09	2.84E-03	7.54E-09	2.84E-03
3a	Small Isolation Failures (liner breach)	1.16E+04	3.93E-07	4.57E-03	3.93E-07	4.57E-03
3b	Large Isolation Failures (liner breach)	1.16E+05	9.82E-08	1.14E-02	9.86E-08	1.15E-02
4	Small Isolation Failures (failure to seal - Type B)	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (failure to seal - Type C)	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	1.41E+06	2.26E-05	3.20E+01	2.26E-05	3.20E+01

Table 5.4 Callaway Annual Dose as a Function of Accident Class Characteristic of Conditions for ILRT Required 3/10 Years (Base Case)						
Accident Classes (Containment Release Type)	**Description**	**Population Dose (person-rem)**	**Without Corrosion**		**With Corrosion [(1)]**	
			Frequency (per Rx-yr)	**Population Dose Rate (person-rem/yr)**	**Frequency (per Rx-yr)**	**Population Dose Rate (person-rem/yr)**
8	Bypass (Interfacing System LOCA)	9.47E+05	5.94E-07	5.62E-01	5.94E-07	5.62E-01
Total			4.27E-05	32.57	4.27E-05	32.57

(1) Only frequencies and releases of Classes 1 and 3b are affected by the corrosion analysis (see Section 4.4).
(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.3 Step 3 - Evaluate Risk Impact of Extending the Type A Test Interval From 10 Years to 15 Years

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to fifteen-years. To do this, an evaluation must first be made of the risk associated with the ten-year interval since the base case applies to a three-year interval (i.e., a simplified representation of a three-in-ten-year interval).

Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the NEI guidance, as described in Section 4.3, by a factor of 3.33 (10 / 3 = 3.33) compared to the base case values. The results of the calculation for a 10-year interval including the corrosion sensitivity case are presented in Table 5.5.

Risk Impact Due to 15-year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 (15 / 3 = 5) compared to the 3-year interval base case value, as described in Section 4.3. The results of the calculation for a 15-year interval including the corrosion sensitivity case are presented in Table 5.6.

Evaluate Change in Total Dose Rate

The population dose (person-rem/yr) is examined to demonstrate the relative change in this parameter. Tables 5.4, 5.5, and 5.6 provide the total doses rate results for the original ILRT interval (3-in-10 years), current ILRT interval (1-in-10 years), and proposed ILRT interval (1-in-15 years), respectively. The change on the total dose rate (DR, person-rem/yr) due to the extension of the ILRT interval can be determined as follow:

$\Delta DR_{15\text{-}10} = DR_{15} - DR_{10}$
= (32.63 person-rem/yr) – (32.61 person-rem/yr)
= 0.02 person-rem/yr

$\Delta DR_{15\text{-}3} = DR_{15} - DR_{3}$
= (32.63 person-rem/yr) – (32.57 person-rem/yr)
= 0.06 person-rem/yr

As stated in Section 1.2, CRITERIA, a small increase in population dose is defined as an increase in population dose of ≤1.0 person-rem per year or 1% of the total population dose, whichever is less restrictive, for the risk impact assessment of the proposed extended ILRT interval. The change in population dose for Callaway is well below the criteria of a small increase in population dose (1.0 person-rem/yr) by extending the Callaway ILRT test interval to 15 years from the original 3-in-10 years and the current 1-in-10 years.

Table 5.5
Callaway Annual Dose as a Function of Accident Class
Characteristic of Conditions for ILRT Required 1/10 Years (Current Interval)

Accident Classes (Containment Release Type)	Description	Population Dose (person-rem)	Without Corrosion		With Corrosion [(1)]	
			Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)	Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)
1	No Containment Failure [(2)]	1.16E+03	1.79E-05	2.08E-02	1.79E-05	2.08E-02
2	Large Isolation Failures (Failure to Close)	3.77E+05	7.54E-09	2.84E-03	7.54E-09	2.84E-03
3a	Small Isolation Failures (liner breach)	1.16E+04	1.31E-06	1.52E-02	1.31E-06	1.52E-02
3b	Large Isolation Failures (liner breach)	1.16E+05	3.27E-07	3.81E-02	3.30E-07	3.84E-02
4	Small Isolation Failures (failure to seal - Type B)	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (failure to seal - Type C)	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A

Table 5.5
Callaway Annual Dose as a Function of Accident Class
Characteristic of Conditions for ILRT Required 1/10 Years (Current Interval)

Accident Classes (Containment Release Type)	Description	Population Dose (person-rem)	Without Corrosion		With Corrosion (1)	
			Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)	Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)
7	Failures Induced by Phenomena (Early and Late)	1.41E+06	2.26E-05	3.20E+01	2.26E-05	3.20E+01
8	Bypass (Interfacing System LOCA)	9.47E+05	5.94E-07	5.62E-01	5.94E-07	5.62E-01
Total			4.27E-05	32.61	4.27E-05	32.61

(1) Only frequencies and releases of Classes 1 and 3b are affected by the corrosion analysis (see Section 4.4).
(2) Characterized as 1L_a release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.6
Callaway Annual Dose as a Function of Accident Class
Characteristic of Conditions for ILRT Required 1/15 Years (Proposed Interval)

Accident Classes (Containment Release Type)	Description	Population Dose (person-rem)	Without Corrosion		With Corrosion (1)	
			Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)	Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)
1	No Containment Failure (2)	1.16E+03	1.71E-05	1.99E-02	1.70E-05	1.98E-02
2	Large Isolation Failures (Failure to Close)	3.77E+05	7.54E-09	2.84E-03	7.54E-09	2.84E-03
3a	Small Isolation Failures (liner breach)	1.16E+04	1.96E-06	2.29E-02	1.96E-06	2.29E-02
3b	Large Isolation Failures (liner breach)	1.16E+05	4.91E-07	5.72E-02	4.96E-07	5.78E-02
4	Small Isolation Failures (failure to seal - Type B)	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (failure to seal - Type C)	N/A	N/A	N/A	N/A	N/A

Table 5.6
Callaway Annual Dose as a Function of Accident Class
Characteristic of Conditions for ILRT Required 1/15 Years (Proposed Interval)

Accident Classes (Containment Release Type)	Description	Population Dose (person-rem)	Without Corrosion		With Corrosion [1]	
			Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)	Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	1.41E+06	2.26E-05	3.20E+01	2.26E-05	3.20E+01
8	Bypass (Interfacing System LOCA)	9.47E+05	5.94E-07	5.62E-01	5.94E-07	5.62E-01
Total			4.27E-05	32.63	4.27E-05	32.63

(1) Only frequencies and releases of Classes 1 and 3b are affected by the corrosion analysis (see Section 4.4).
(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.4 Step 4 - Determine the Change in Risk in Terms of Large Early Release Frequency (LERF)

The risk increase 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 an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the NEI guidance, 100% of the Class 3b contribution would be considered LERF.

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 CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Callaway, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the NEI methodology). Based on a ten-year test interval from Table 5.5, the Class 3b frequency with corrosion is 3.30E-07/yr; and based on a fifteen-year test interval from Table 5.6, it is 4.96E-07/yr, with corrosion. Thus the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 10 to 15 years is:

$$\Delta LERF_{15\text{-}10} = LERF_{class3b\ 15} - LERF_{class3b\ 10}$$
$$= (4.96E\text{-}07/yr) - (3.30E\text{-}07/yr)$$
$$= 1.66E\text{-}07/yr$$

Similarly, the increase due to increasing the interval from 3 to 15 years is:

$$\Delta LERF_{15\text{-}3} = LERF_{class3b\ 15} - LERF_{class3b\ 3}$$
$$= (4.96E\text{-}07/yr) - (9.86E\text{-}08/yr)$$
$$= 3.98E\text{-}07/yr$$

Hence, even with the conservatisms included in the evaluation (per the NEI methodology), the estimated change in LERF is just above the threshold criteria for a very small change when comparing the 15-year results to the current 10-year requirement. It is above the criteria for a very small change but well below the criteria for a small change when compared to the original 3-year requirement.

The total LERF from the EPRI methodology is comprised of those sequences from Class 3b, Class 2, and Class 8. The total LERF for the original interval (3 years), current interval (10 years), and proposed interval (15 years) with corrosion can thus be determined as follows:

$$LERF_3 = (F_{class\ 2} + F_{class\ 3b} + F_{class\ 8})_{3yrs}$$
$$= (7.54E\text{-}09/yr) + (9.86E\text{-}08/yr) + (5.94E\text{-}07/yr)$$
$$= 7.00E\text{-}07/yr$$

$$LERF_{10} = (F_{class\ 2} + F_{class\ 3b} + F_{class\ 8})_{10yrs}$$
$$= (7.54E\text{-}09/yr) + (3.30E\text{-}07/yr) + (5.94E\text{-}07/yr)$$
$$= 9.31E\text{-}07/yr$$

$$LERF_{15} = (F_{class\ 2} + F_{class\ 3b} + F_{class\ 8})_{15yrs}$$
$$= (7.54E\text{-}09/yr) + (4.96E\text{-}07/yr) + (5.94E\text{-}07/yr)$$
$$= 1.10E\text{-}06/yr$$

5.5 Step 5 - Determine the Impact on the Conditional Containment Failure Probability (CCFP)

Another parameter that can provide input into the decision-making process, per the NRC guidance in RG 1.174, is the change in the CCFP. The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the EPRI Report No. 1009325, Revision 2-A [30]. The NRC has previously accepted similar calculations [12] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy.

$$CCFP = [1 - (\text{Class 1 Frequency} + \text{Class 3a Frequency}) / CDF] * 100\%$$

$$CCFP_3 = [1 - (1.90E\text{-}05/yr + 3.93E\text{-}07/yr) / (4.27E\text{-}05/yr)] * 100\%$$
$$= 54.55\%$$

$$CCFP_{10} = [1 - (1.79E\text{-}05/yr + 1.31E\text{-}06/yr) / (4.27E\text{-}05/yr)] * 100\%$$
$$= 55.09\%$$

$$CCFP_{15} = [1 - (1.70E\text{-}05/yr + 1.96E\text{-}06/yr) / (4.27E\text{-}05/yr)] * 100\%$$
$$= 55.48\%$$

$$\Delta CCFP_{15\text{-}10} = CCFP_{15} - CCFP_{10}$$
$$= 55.48\% - 55.09\%$$

$= 0.39\%$

$$\Delta CCFP_{15\text{-}3} = CCFP_{15} - CCFP_3$$
$$= 55.48\% - 54.55\%$$
$$= 0.93\%$$

As stated in Section 1.2, CRITERIA, a small increase in CCFP is defined as an increase in CCFP ≤1.5 percentage point for the risk impact assessment of the proposed extended ILRT interval. The change in CCFP is less than 1% in this assessment by extending the Callaway ILRT test interval to 15 years from the original 3-in-10 years and the current 1-in-10 years.

5.6 Summary of Results

The results from this ILRT extension risk assessment for Callaway are summarized in Table 5.7 (including the age adjusted steel liner corrosion likelihood) and Table 5.8 (excluding the age adjusted steel liner corrosion). As can be seen from the two tables, the LERF for a 15-year Type A test interval with corrosion is 1.10E-06/yr while the LERF for the same interval without corrosion is 1.09E-06/yr. The change in LERF between the corrosion and non-corrosion cases for a 15-year test interval is very small, 1E-08/yr. Since the LERF for the base case, 3 tests per 10 years, is the same, 7.00E-07/yr, for both corrosion and non-corrosion cases, the difference in delta-LERF between the corrosion and non-corrosion cases is also very small. Similarly, the differences in CCFP and population dose rate between the corrosion and non-corrosion cases are very small.

Table 5.7
Callaway ILRT Risk Assessment: Base Case, Current Interval, and Proposed Interval (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	Dose (person-rem)	Base Case (3 in 10 years)		Current Interval (1 in 10 years)		Proposed Interval (1 in 15 years)	
		Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)	Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)	Frequency (per Rx-yr)	Population Dose Rate (person-rem/yr)
1	1.16E+03	1.90E-05	2.21E-02	1.79E-05	2.08E-02	1.70E-05	1.98E-02
2	3.77E+05	7.54E-09	2.84E-03	7.54E-09	2.84E-03	7.54E-09	2.84E-03
3a	1.16E+04	3.93E-07	4.57E-03	1.31E-06	1.52E-02	1.96E-06	2.29E-02
3b	1.16E+05	9.86E-08	1.15E-02	3.30E-07	3.84E-02	4.96E-07	5.78E-02
7	1.41E+06	2.26E-05	3.20E+01	2.26E-05	3.20E+01	2.26E-05	3.20E+01
8	9.47E+05	5.94E-07	5.62E-01	5.94E-07	5.62E-01	5.94E-07	5.62E-01
Total		4.27E-05	32.57	4.27E-05	32.61	4.27E-05	32.63

Table 5.7
Callaway ILRT Risk Assessment: Base Case, Current Interval, and Proposed Interval (Including Age Adjusted Steel Liner Corrosion Likelihood)

Delta Total Dose Rate (person-rem/yr)	From 3 yr	N/A	0.04	0.06
	From 10 yr	N/A	N/A	0.02
LERF (per year)		7.00E-07	9.31E-07	1.10E-06
ΔLERF (per year)	From 3 yr	N/A	2.31E-07	3.98E-07
	From 10 yr	N/A	N/A	1.66E-07
CCFP		54.55%	55.09%	55.48%
ΔCCFP	From 3 yr	N/A	0.54%	0.93%
	From 10 yr	N/A	N/A	0.39%

Table 5.8
Callaway ILRT Risk Assessment: Base Case, Current Interval, and Proposed Interval (Excluding Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	**Dose (person-rem)**	**Base Case (3 in 10 years)**		**Current Interval (1 in 10 years)**		**Proposed Interval (1 in 15 years)**	
		Frequency (per Rx-yr)	**Population Dose Rate (person-rem/yr)**	**Frequency (per Rx-yr)**	**Population Dose Rate (person-rem/yr)**	**Frequency (per Rx-yr)**	**Population Dose Rate (person-rem/yr)**
1	1.16E+03	1.90E-05	2.21E-02	1.79E-05	2.08E-02	1.71E-05	1.99E-02
2	3.77E+05	7.54E-09	2.84E-03	7.54E-09	2.84E-03	7.54E-09	2.84E-03
3a	1.16E+04	3.93E-07	4.57E-03	1.31E-06	1.52E-02	1.96E-06	2.29E-02
3b	1.16E+05	9.82E-08	1.14E-02	3.27-07	3.81E-02	4.91E-07	5.72E-02
7	1.41E+06	2.26E-05	3.20E+01	2.26E-05	3.20E+01	2.26E-05	3.20E+01
8	9.47E+05	5.94E-07	5.62E-01	5.94E-07	5.62E-01	5.94E-07	5.62E-01
Total		4.27E-05	32.57	4.27E-05	32.61	4.27E-05	32.63

Table 5.8
Callaway ILRT Risk Assessment: Base Case, Current Interval, and Proposed Interval (Excluding Age Adjusted Steel Liner Corrosion Likelihood)

Delta Total Dose Rate (person-rem/yr)	From 3 yr	N/A	0.04	0.06
	From 10 yr	N/A	N/A	0.02
LERF (per year)		7.00E-07	9.29E-07	1.09E-06
ΔLERF (per year)	From 3 yr	N/A	2.29E-07	3.93E-07
	From 10 yr	N/A	N/A	1.64E-07
CCFP		54.55%	55.09%	55.47%
ΔCCFP	From 3 yr	N/A	0.54%	0.92%
	From 10 yr	N/A	N/A	0.38%

6.0 SENSITIVITIES

6.1 Potential Impact from External Events Contribution

As stated in Section 3.0, GROUND RULES, this analysis uses the representative results provided in the Callaway Level 1 and Level 2 internal events PRA model. It is believed to be appropriate to use the Callaway internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. The analysis results presented in Section 5.0, RESULTS, are based on Callaway's internal events PRA model. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increase in population dose) would not substantially differ if fire, seismic and other external events (including internal flooding) were to be included in the calculations. This is consistent with the guidance in EPRI Report No. 1009325.

Currently Callaway has no quality external event analysis that could be used to apply the EPRI methodology to assess the contribution of the external events (e.g., fire and seismic) in this ILRT risk impact assessment. Similarly, Callaway internal flooding was analyzed using a successive screening process and no Level 2 analysis has been performed for internal flooding scenarios. Thus, an alternate method was used to assess an order-of-magnitude estimate for contribution of the external events, including internal flooding, to the impact of the changed interval.

In this analysis (Section 5.0), the total internal event LERF for the base line is determined to be 7.00E-07/yr with the conservative EPRI methodology. It is likely that the total external event LERF would be less than the internal event LERF given that some LERF events such as Interfacing System Loss of Coolant Accidents (ISLOCA) and Steam Generator Tube Ruptures (SGTR) which contribute directly to LERF are not initiated from external events. Conservatively assuming the LERF for external events to be equal to that of internal events, the total LERF from both internal events and external events can be calculated by using the results from Table 5.7 which include the age adjusted steel liner corrosion likelihood.

$$\begin{aligned} LERF_3 \text{ (internal + external)} &= 2 * LERF_3 \text{ (internal)} \\ &= 2 * (7.00E\text{-}07/yr) \\ &= 1.40E\text{-}06/yr \end{aligned}$$

$$\begin{aligned} LERF_{10} \text{ (internal + external)} &= 2 * LERF_{10} \text{ (internal)} \\ &= 2 * (9.31E\text{-}07/yr) \\ &= 1.86E\text{-}06/yr \end{aligned}$$

$$\begin{aligned} LERF_{15} \text{ (internal + external)} &= 2 * LERF_{15} \text{ (internal)} \\ &= 2 * (1.10E\text{-}06/yr) \\ &= 2.20E\text{-}06/yr \end{aligned}$$

The total increase in the overall probability of LERF from both internal events and external events due to increasing the ILRT test interval from 10 to 15 years is:

$$\begin{aligned} \Delta LERF_{15\text{-}10} \text{ (internal + external)} &= (LERF_{15} - LERF_{10}) \text{ (internal + external)} \\ &= (2.20E\text{-}06/yr) - (1.86E\text{-}06/yr) \\ &= 3.4E\text{-}07/yr \end{aligned}$$

Similarly, the increase due to increasing the interval from 3 to 15 years is:

$$\begin{aligned} \Delta LERF_{15\text{-}3} \text{ (internal + external)} &= (LERF_{15} - LERF_{3}) \text{ (internal + external)} \\ &= (2.20E\text{-}06/yr) - (1.40E\text{-}06/yr) \\ &= 8.0E\text{-}07/yr \end{aligned}$$

As can be seen, with the conservative assumption that LERF for external events is equal to that of internal events, the estimated change in LERF for is still below the criteria for a small change when comparing the 15 year results to the current 10-year requirement or to the original 3-year requirement. The total LERF for both internal events and external events with the Type A test interval of 15 years is well below the RG 1.174 acceptance guideline of 1E-05 per year.

6.2 Sensitivity to Class 3b Contribution to LERF

The Class 3b frequency for the base case of a three in ten-year ILRT interval with corrosion is 9.86E-08/yr [Table 5.4]. Extending the interval to one in ten years results in a frequency of 3.30E-07/yr [Table 5.5]. Extending it to one in fifteen years results in a frequency of 4.96E-07/yr [Table 5.6], which is an increase of 1.66E-07/yr from the current ILRT interval (one in ten years) and 3.97E-07/yr from the base case (three in ten years). If 100% of the Class 3b sequences are assumed to have potential release large enough for LERF, then the increase in LERF due to extending the interval from three in ten to one in fifteen is still below the RG 1.174 threshold for small changes in LERF of 1E-06/yr.

6.3 Sensitivity to Corrosion Impact Assumptions

The results in Tables 5.7 and 5.8 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the cylinder and dome and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6-1. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 1.48E-07/yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6.1
Callaway Steel Liner Corrosion Sensitivity Cases

Age (Step 2&3 in the corrosion analysis)	**Containment Breach Likelihood (Step 4 in the corrosion analysis)**	**Visual Inspection Failure (Step 5 in the corrosion analysis)**	**Increase in Class 3b Frequency (ΔLERF) for ILRT Extension from 3 to 15 Years (per Rx-yr)**	
			Total Increase	**Increase Due to Corrosion**
Base Case *Double every 5 yrs*	Base Case *1% Cylinder & 0.1% Basemat*	Base Case *10%*	3.98E-07	4.74E-09
Double every 2 yrs	Base Case	Base Case	4.03E-07	1.06E-08
Double every 10 yrs	Base Case	Base Case	3.97E-07	3.89E-09
Base Case	Base Case	*15%*	3.99E-07	6.49E-09
Base Case	Base Case	*5%*	3.96E-07	2.78E-09
Base Case	*10% Cylinder & 1% Basemat*	Base Case	4.39E-07	4.64E-08
Base Case	*0.1% Cylinder & 0.01% Basemat*	Base Case	3.93E-07	4.64E-10
Lower Bound				
Double every 10 yrs	*0.1% Cylinder & 0.01% Basemat*	*5%*	3.93E-07	2.32E-10
Upper Bound				
Double every 2 yrs	*10% Cylinder & 1% Basemat*	*15%*	5.41E-07	1.48E-07

7.0 CONCLUSION

Based on the results from Section 5 and the sensitivity calculations presented in Section 6, the conclusions regarding the assessment of the plant risk associated with extending the Type A ILRT test frequency to fifteen years are as follows:

- RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines small changes in risk as resulting in increases of CDF below 10^{-5}/yr and increases in LERF below 10^{-6}/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 one in ten years (current interval) to one in fifteen years (proposed interval) is conservatively estimated as 1.66E-07/yr using the NEI guidance. The increase in LERF resulting from three in ten years (original interval) to one in fifteen years (proposed interval) is conservatively estimated as 3.98E-07/yr. As such, the estimated change in LERF is determined to be small using the acceptance guidelines of RG 1.174.
- RG 1.174 also states that when the calculated increase in LERF is in the range of 1.0E-6 per reactor year to 1.0E-7 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-5 per reactor year. An additional estimation of the impact from external events was also made. In this case, the total LERF was conservatively estimated as 2.20E-06 per reactor year for Callaway. This is well below the RG 1.174 acceptance criterion for total LERF of 1.0E-5.
- The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.02 person-rem/yr when compared with the current Type A test frequency (once-per-ten-years), and 0.06 person-rem/yr when compared with the original Type A test frequency (three-in-ten-years). The NRC Final Safety Evaluation [7] and EPRI Report No. 1009325, Revision 2-A [30], state that a small increase in population dose is defined as an increase of $\leq$ 1.0 person-rem per year or $\leq$ 1% of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The risk increase of this change is well below the above acceptance criterion and therefore is determined to be small.
- The increase in the conditional containment failure frequency from the one-in-ten-year interval (current interval) to a one-in-fifteen-year interval (proposed interval) is 0.39%. The increase from the three-in-ten-year interval (original interval) to a one-in-fifteen-year interval is 0.93%. The NRC Final Safety Evaluation and EPRI Report No. 1009325, Revision 2-A, state that increases in CCFP of $\leq$ 1.5 percentage point are small for the risk impact assessment of the extended ILRT intervals. As such, this increase is found to be small.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant as it represents a small change to the Callaway risk profile. This conclusion is also consistent with those results from the previous assessments. The NRC concluded in NUREG-1493 that:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for Callaway confirm these general findings on a plant specific basis considering the severe accidents evaluated for Callaway, the Callaway containment failure modes, and the local population surrounding the Callaway Plant.

8.0 REFERENCES

[1] *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, NEI 94-01, Revision 2, August 2007.

[2] *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, EPRI, Palo Alto, CA, Report No. 1009325, Revision 2, August 2007.

[3] *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, July 1998.

[4] *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, EPRI, Palo Alto, CA, EPRI TR-104285, Final Report, August 1994.

[5] *Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.

[6] *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension*, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.

[7] *Final Safety Evaluation for NEI Topical Report (TR) 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2 (TAC No. MC9663)*, U.S. Nuclear Regulatory Commission, Project No. 689, June 25, 2008.

[8] *Performance-Based Containment Leak-Test Program*, NUREG-1493, September 1995.

[9] *Individual Plant Examination (IPE) Report for the Callaway Plant*, Revision 5, Callaway Plant, June 2006.

[10] *Evaluation of Severe Accident Risks: Surry Unit 1*, Main Report NUREG/CR-4551, SAND86-1309, Volume 3, Revision 1, Part 1, October 1990.

[11] Letter from R. J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.

[12] U.S. Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.

[13] *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.

[14] *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.

[15] *Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check'*, NUREG-1273, April 1988.

[16] *Review of Light Water Reactor Regulatory Requirement*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986.

[17] *Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™*, EPRI, Palo Alto, CA, TR-105189, Final Report, May 1995.

[18] *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, NUREG-1150, December 1990.

[19] U.S. Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.

[20] Anthony R. Pietrangelo, *One-Time Extensions of Containment Integrated Leak Rate Test Interval – Additional Information*, NEI letter to Administrative Points of Contact, November 30, 2001.

[21] Letter from J. A. Hutton (Exelon, Peach Bottom) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, May 30, 2001.

[22] *Risk Assessment for Joseph M. Farley Nuclear Plant Regarding ILRT (Type A) Extension Request*, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, P0293010002-1929-030602, March 2002.

[23] Letter from D. E. Young (Florida Power, Crystal River) to U.S. Nuclear Regulatory Commission, 3F0401-11, April 25, 2001.

[24] *Risk Assessment for Vogtle Electric Generating Plant Regarding the ILRT (Type A) Extension Request*, Prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, February 2003.

[25] *Individual Plant Examination*, Revision 5 with the Fourth Update of the Callaway Level 1 PRA, Callaway Plant, June 2006.

[26] *Updated Final Safety Analysis Report*, Callaway Plant.

[27] *ESP-SM-01001, Containment Leakage Rate Testing Program*, Revision 022, Callaway Plant, May 2008.

[28] *ESP-GP-01007, Reactor Building Integrated Leak Rate Test*, Revision 012, Callaway Plant, May 2008.

[29] *PRAER 08-322, ILRT Extension Task 3 Evaluation*, Revision 0, Callaway Plant, August 2008.

[30] *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, EPRI, Palo Alto, CA, Report No. 1009325, Revision 2-A, October 2008.

ULNRC-05598

ATTACHMENT 5

CALLAWAY PRA TECHNICAL ADEQUACY REPORT FOR THE ILRT (TYPE A) INTERVAL EXTENSION REQUEST

CALLAWAY PRA TECHNICAL ADEQUACY REPORT FOR THE ILRT (TYPE A) INTERVAL EXTENSION REQUEST

1.0 INTRODUCTION

The risk-informed support for the proposed integrated leak rate test (ILRT) interval change is based in part upon the analyses that include results from the Callaway Level 1 and Level 2 Probabilistic Risk Assessment (PRA) models. The Callaway PRA needs to be technically adequate to support the license amendment request for ILRT interval extension.

As stated in the NRC Final Safety Evaluation [1] for NEI 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and EPRI Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Capability Category I of ASME RA-Sa-2003 shall be applied as the standard to address the technical adequacy of the PRA model since "approximate values of CDF and LERF and their distribution among release categories are sufficient for use in the EPRI methodology."

This risk assessment for Callaway ILRT (Type A) interval extension request follows the guidelines from NEI 94-01, Revision 2 [2] and the methodology used in EPRI Report No. 1009325, Revision 2 [3]. Capability Category I of ASME RA-Sa-2003 will be applied as the standard for the application. Any identified deficiencies in addressing this standard will be assessed in the following section.

2.0 TECHNICAL ADEQUACY ASSESSMENT

2.1 Callaway PRA Gap Analysis

A gap analysis has been performed on Callaway PRA to evaluate its level of compliance with Regulatory Guide (RG) 1.200. This independent assessment reviewed the entire current Callaway PRA model, which is composed of the Individual Plant Examination (IPE), Union Electric calculation packages and Addenda, the Individual Plant External Event Examination (IPEEE), and the low power and shutdown safety monitor model, against the applicable standards. The analysis identified the areas of the PRA which need to be strengthened in order to assure that the Callaway PRA conforms to the Category II requirements of the ASME PRA Standard updated to include Addenda B, the Category II requirements of ANSI/ANS-58.21-2003, and the expected requirements of the ANS Low Power and Shutdown PRA Standard. Callaway PRA Gap Analysis Report [4] documents the results of the gap analysis. Table 2.1.1 provides an overview of the results of the analysis for the internal events during full power, with the number of supporting requirements (SRs) that are meeting or not meeting the various Capability Category II requirements for each PRA element.

Table 2.1.1 Callaway PRA Category II SRs Status -- Internal Events, Full Power				
PRA Element	**SRs**	**Cat II / Met**	**Not Met**	**Not Applicable**
IE	35	17	14	4
AS	21	16	4	1
SC	15	11	4	0
SY	44	39	2	3
HR	36	32	3	1
DA	33	25	8	0
IF	54	37	9	8
QU	36	27	8	1
LE	42	36	6	0
MU	10	8	2	0
Total	326	248	60	18

Any requirements of the Standard that were believed to not be fully complied with were noted in Finding/Observation (F&O) format. Table 2.1.2 provides the number of F&Os that were identified for each PRA element with different levels of significance.

Table 2.1.2 Callaway PRA HLR F&Os Summary			
HLR	**Total F&Os**	**Level A/B**	**Level C**
IE	14	5	9
AS	7	6	1
SC	4	0	4
SY	3	2	1
HR	3	0	3
DA	3	1	2
IF	6	4	2
QU	12	6	6
LE	3	2	1
MU	2	0	2
Total	57	26	31

None of the "A/B" F&Os were identified as "A" level, i.e. extremely important and necessary to address in order to assure the technical adequacy of the PRA or the quality of the PRA or the quality of the PRA update process. All "A/B" items were identified on the F&O forms as "B", which means that the findings need to be addressed but do not cause the PRA to be technically inadequate from an overall perspective. "C" F&Os are considered desirable to maintain maximum flexibility in PRA applications and consistency in the industry. Since "C" F&Os are only of marginal importance, they would have negligible impact on the risk assessment and will not be further assessed.

Table 2.1.3 presents the justification of the twenty-six "A/B" F&Os identified in the Gap Analysis for their impacts on the risk assessment for Callaway's ILRT interval extension request. These open items would not impact the results of the risk assessment for the ILRT extension.

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
1	IE-6	B	Open	There was no evidence found that operating experience was reviewed with precursors in mind. If an event did not result in the generation of a trip or an LER, then it was not reviewed. Interviews with operations and maintenance personnel would be one method to meet SR IE-A7. The current analysis does not meet Cat 2 SR IE-A7.	FMEAs were used to determine the plant impact of failures for important support systems. Precursors found from operating experience should only confirm the results of the FMEAs (i.e., that failures of certain equipment can result in an initiator and failure of mitigating equipment), and should not change the results obtained from the FMEAs. The inclusion of precursor events could possibly impact the initiating event frequencies for some non-support system initiators (e.g., reactor trip). However, it is unlikely enough undiscovered precursors exist to significantly affect these frequencies. The Callaway internal events and their frequencies were compared to those of the industry. The results indicate no outliers for Callaway. Therefore, this F/O should not impact the results of the PRA evaluation for the one-time ILRT extension.
2	IE-7	B	Open	The IE frequencies currently do not include any uncertainty bounds. The IE frequencies need to have uncertainty bounds assigned to meet SR IE-C1, IE-C1a, and IE-C13.	Uncertainty bounds on the IE frequencies do not impact the point estimates of CDF or LERF. Including uncertainty bounds on the IE frequencies would likely change an uncertainty analysis distribution. However, it is unlikely that such a change would preclude an application. Therefore, this F/O should not impact the results of the PRA evaluation for the one-time ILRT extension.

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
3	IE-8	B	Open	The Callaway PRA credits repair of hardware faults in the recovery of the loss of CCW and loss of SWS initiating events. The repair events, which include repair of CCF of pumps and valves, lack sufficient analysis or data. Crediting repair of components is not acceptable unless the probability of repair is justified through an adequate analysis or examination of data. The Callaway PRA does not meet SR IE-C1b, IE-C9, and SY-A22.	The basic events associated with these hardware recoveries are risk-insignificant in the baseline Callaway PRA model. The loss of CCW and loss of SWS initiating events have minimal impact on LERF. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.
4	IE-12	B	Open	There is no documentation of a comparison with generic data sources for the support system initiating event fault tree results. This comparison needs to be documented as part of each update in order to meet SR IE-C10.	The Callaway support system IE frequencies were compared to those of other WOG plants, using information in a WOG database, for an early Callaway PRA update. The frequencies were consistent. In addition, the cooling water support system initiators were compared to other WOG plants in WCAP-16464-NP, for MSPI purposes. Callaway was not identified as an outlier. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.
5	IE-13	B	Open	The Callaway treatment of ISLOCA addresses items a-d and may include item e but that is not clear. The ISLOCA documentation is good for the evaluation of the high/low interfaces (ZZ-105) however the documentation of the quantification from that point on is minimal, is not incorporated in the main model, and has not been revised or re-examined since the IPE submittal. The ISLOCA model as it now stands does not meet SR IE-C12.	In the Callaway PRA, an ISLOCA event is assumed to result in core damage and large early release. No mitigation is credited. While ISLOCA is a major contributor to LERF, an ISLOCA event is not impacted by containment isolation capability. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
6	AS-1	B	Open	Event Tree T(SW), function L2SW-M should evaluate the TDAFW pump with no functioning SW/ESW equipment. The cutsets for this function include failures of the ESW pumps and human action failures for alignment of SW/ESW. Since the initiator fails all SW/ESW, the logic should not include these events. A similar situation exists for function L2T1s. Event Tree T(SW) function O1SW-M includes a FANDB operator error which does not belong in the function. A similar situation exists for functions O1C-M, O1CT1-M, and O1SW-M.	Correction of these functions, which also addresses similar issues in F/Os AS-3, AS-7, SY-1, QU-3, and QU-4, would result in a small increase in the total baseline CDF (approx. 1%) for the Callaway PRA. The T(SW) and T(1S) events have an insignificant contribution to LERF. Therefore, these F/Os would not impact the results of the PRA evaluation for the one-time ILRT extension.
7	AS-2	B	Open	Transfers between event trees may be used to reduce the size and complexity of individual event trees. DEFINE any transfers that are used and the method that is used to implement them in the qualitative definition of accident sequences and in their quantification. USE a method for implementing an event tree transfer that preserves the dependencies that are part of the transferred sequence. These include functional, system, initiating event, operator, and spatial or environmental dependencies. This requirement is not met. Many transfers such as seal LOCA and stuck open PORV transfer to a "psuedo event tree". These transfers are quantified using an OCL file that does not have a specific event tree. This introduces possibilities for error in the quantification since there is no event tree on which to base the evaluated functions, especially those that require preservation of dependencies. The actual event tree for quantification of the RCP seal LOCA events was not found. An event tree T(RCP) appears to have been used, but this event tree has an event	This is a documentation issue. The transfer sequences have been extensively reviewed and no issues have been identified. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
				for recovery of CCW, which is not included in the .OCL files for the RCP seal LOCA events.	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
8	AS-3	B	Open	The method of event tree analysis for support system initiators does not appear to correctly capture the failed dependencies in the mitigating systems for some support system IE's. A single basic event is used for the initiating event. House events are included in the fault trees to turn off the affected trains when a support system is not available. It is not clear there are sufficient support systems modeled in the main feedwater and non-safety service water to fail these systems when their support systems are unavailable. This may occur in T(SW), T(NK01), and T(NK04). The cutsets for T(SW), T(NK01), T(NK04), and T(CCW) should be checked to search for systems that would be failed by the loss of the initiator, and then modify the fault trees to include the appropriate house events to disable these systems.	Review of the support system initiators reveals that this F/O actually only pertains to T(SW). This issue was addressed in the response to F/O AS-1, above.
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
9	AS-4	B	Open	The RCP seal LOCA model needs to be updated to reflect the latest WOG model, which is approved by the NRC.	The current Callaway PRA model utilizes the RCP seal LOCA model of WCAP-10541, in which the 21 gpm/pump seal LOCA has a probability of occurrence of approximately 90%. The WOG2000 RCP seal LOCA model (documented in WCAP-15603), uses a probability of approximately 80% for the 21 gpm/pump seal LOCA. A sensitivity analysis was performed to address this source of uncertainty related to seal LOCA. The associated core uncovery probabilities, following loss of RCP seal cooling, were increased by 25 percent to approximate the impact of the WOG2000 RCP seal failure probabilities, and resulted in an insignificant increase of approximately 1.5% in CDF. The initiating events impacted by the RCP seal LOCA model

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
					have minimal impact on LERF. Therefore, this F/O would have minimal impact on the results of the PRA evaluation for the one-time ILRT extension.
10	AS-6	B	Open	The MAAP results indicate there are 60 hours before core melt for the SGTR sequence with failure to isolate the SG. If the MAAP analysis is correct, then the sequence should be screened. If the MAAP analysis is not correct, or MAAP 3 can not provide a correct representation of the sequence, MAAP 4 should be used.	Retaining these sequences in the results is slightly conservative for CDF (approx. 1% of CDF). Retaining these sequences is very conservative for LERF (approx. 70% of LERF), although LERF remains in the E-7 range. Since these are unisolated SGTR sequences, they are not impacted by containment isolation capability. Therefore, this F/O would have minimal impact on the results of the PRA evaluation for the one-time ILRT extension.
11	AS-7	B	Open	Specific errors are as noted below: Function O1T1S in the SBO event tree contains basic events for MFW and SW as a backup source for water to SGs if the TDP fails. The problem occurs in the SECDEP fault tree, which asks for GMFX100, but does not have any logic to cancel the gate in SBO. There are no events in the MFX fault tree which will cancel it in the event of an SBO, either. Also, in MFW.lgc, gate GMFW413 – the SVC system will be failed by LOSP, but comes through the link in the SBO function. Back-up sources of water to the SG are modeled at a high level, often only represented by an HEP. There needs to be either a) support systems developed which will be failed by LOSP or AC power, or b) house event logic to fail these for SBO. The AFW function on the T(SW) event tree – (L2SW-M) – has recovery factors for ESW as a suction source to the turbine driven AFW pump. (AL-XHE-FO-AFWESW). ESW is failed by the initiator, but the IE is a basic event, not cutsets. Need to represent the initiator as a support system fault tree, OR need	This issue was addressed in the response to F/O AS-1, above.

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
				to include house events in the AFW function to fail the cross-tie to the ESW system after a Loss of ESW. In T(SW) event tree, function O1SW-M has an event (AE-XHE-FO-MFWFLO) for failure of MFW as back up to AFW. MFW is unavailable after loss of SW. Need to include support systems for MFW or insert house events in fault tree to turn off MFW for loss of T(SW).	
12	SY-1	B	Open	For the Instrument Air System a single basic event is used and is based on generic data. The Callaway plant is not highly dependent upon IAS and the PRA loads on IAS also are supplied with N2 backup which is modeled. Modeling the IAS as a single basic is acceptable however, the MFW dependency on the IAS is not modeled and needs to be included since MFW is credited as a backup to AFW and is important. The actuation system is modeled with a single event for every redundancy which is set to fail for scenarios in which the conditions are not present to generate the signal. The level of detail is acceptable for this use. The dependency of MFW on IAS needs to be included and the data associated with these single event failures need to be reviewed against current industry data and updated if necessary. The applicability of the data to the Callaway configuration also needs to be justified. One such source of data is NUREG/CR-5750.	The IAS consists of three compressors, two of which are cooled by ESW and one that is cooled by NSW. Parts of the MFW system and the condensate system are dependent on IAS. MFW and IAS are part of the modeled PRA function to cooldown and depressurize the RCS. This action occurs with successful secondary side cooling but failed primary high head injection for events with a primary leak. This dependency between ESW/NSW, IAS, and MFW has an insignificant affect on the PRA results, except for the T(SW) event. For this event, this issue was addressed in the response to F/O AS-1, above. Note that safety-related components using instrument air also have safety-related nitrogen accumulators to support their operation.

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
13	SY-2	B	Open	The Callaway PRA adequately models CCFs with the exception of battery chargers and breakers as noted in SR SY-B1 and B3. The quantification of all CCFs should be updated. CCFs should be added for Battery Chargers and Breakers. The quantification of the CCFs should be done in accordance with NUREG/CR-5485.	The Battery Charger basic events are not risk significant in the Callaway PRA model. A Battery Charger CCF basic event is not expected to be risk significant. Many of the breaker basic events are risk significant, so a breaker CCF basic event would also be expected to be risk significant and would probably slightly increase the baseline total CDF. However, only the SGTR event in the baseline Callaway PRA would have a significant impact on LERF. A review of the cutsets for the SGTR event indicates that breaker CCF should have an insignificant impact on the SGTR results. Therefore, this F/O should not impact the results of the PRA evaluation for the one-time ILRT extension.
14	DA-2	B	Open	Group estimations are based only on component type. Capability Category II requires grouping of components according to type (e.g., motor-operated pump, air-operated valve) and according to the characteristics of their usage to the extent supported by data: (a) mission type (e.g., standby, operating) (b) service condition (e.g., clean vs. untreated water, air) The level of grouping used in the latest data update uses a very fine grouping which leads to a smaller data pool for each different component. Consideration should be given to collecting data on as large a group of components as possible to establish a meaningful collection of data. Grouping of the components as defined in SR DA-B1 and DA-B2 provides a more reasonable aggregation of data and results in a larger data pool to characterize the failure data.	A more recent data update performed to the ASME standard, grouped pumps and valves by component type, service conditions, etc. The resulting groupings had populations that were similar to the groupings that are the subject of this F/O. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.
15	IF-2	B	Open	This requirement is not met at any Category. The Category I/II screening quantitative criterion in the standard is 1E-09/year. ZZ-466 screening criterion was 1E-06/yr.	Flooding was addressed explicitly in the one-time ILRT extension.

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
16	IF-4	B	Open	If additional human failure events are required to support quantification of flood scenarios, PERFORM any human reliability analysis in accordance with the applicable requirements described in Tables 4.5.5-2(e) through Table 4.5.5-2(h). This requirement is not met. The HEP values used in ZZ-466 are not developed from a human reliability analysis.	Flooding was addressed explicitly in the one-time ILRT extension.
17	IF-5	B	Open	For each defined flood area and each flood source, IDENTIFY those automatic or operator responses that have the ability to terminate or contain the flood propagation. This requirement is not met. ZZ-466 treats operator response in a generic sense.	Flooding was addressed explicitly in the one-time ILRT extension.
18	IF-6	B	Open	For each flood scenario, REVIEW the LERF analysis to confirm applicability of the LERF sequences. If appropriate LERF sequences do not exist, MODIFY the LERF analysis as necessary to account for any unique flood-induced scenarios or phenomena in accordance with the applicable requirements described in para. 4.5.9. This requirement is not met. The internal flooding sequences are not considered in the LERF analysis.	Flooding was addressed explicitly in the one-time ILRT extension.

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
19	QU-1	B	Open	The current quantification does not include an uncertainty calculation to account for the "state-of-knowledge" correlation between event probabilities. The structure exists to perform this correlation within WinNUPRA but at the current time it has not been done.	The "state-of-knowledge" correlation generally pertains to the data applied to equipment across trains. For example, an SBO cutset may contain the failure of the "A" and "B" EDGs. The failure data for both EDGs most likely is based on the same source of information. Therefore, any uncertainty analysis should vary the failure data for these components in the same manner (i.e., the data is not independent). The "state-of-knowledge" correlation would only impact an uncertainty analysis. However, it is unlikely that an uncertainty analysis that addresses the "state-of-knowledge" correlation would preclude an application. Therefore, this F/O should not impact the results of the PRA evaluation for the one-time ILRT extension.
20	QU-3	B	Open	Some instances of incorrect transfer of sequence characteristics, incorrect logic, incorrect house event settings, and resultant cutsets were identified based on cutset reviews. The process is generally set up correctly but the overall process would benefit from revising the quantification process to account for the additional software capability currently available. As a minimum, the top cutsets (500?) need to be reviewed to make sure that the transfers, logic, house event setting are yielding realistic combinations.	The instances of incorrect transfer of sequence characteristics, incorrect logic, etc., were specifically identified in F/O AS-1. The response to F/O AS-1, above, estimated the impact of these instances on CDF and LERF.
21	QU-4	B	Open	The IAS is correctly failed for LOSP, but remains available in all other cases. The IAS is cooled by SW and would be unavailable after loss of all SW (T(SW)) and should be set to failed via a house event setting. The availability of IAS needs to be propagated correctly during the quantification process.	IAS modeling problems were specifically identified in F/O SY-1. The response to F/O AS-1, above, estimated the impact of these IAS issues on CDF and LERF.

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
22	QU-9	B	Open	In general the model integration process is adequately documented; however, several of the areas do not meet the requirements. Items b (records of the cutset review process), f (the accident sequences and their contributing cutsets), g (equipment or human actions that are the key factors in causing the accidents to be non-dominant), and i (the uncertainty distribution for the total CDF) are not addressed in the documentation. As a minimum, these items need to be addressed to meet SR QU-F2. If the quantification process and documentation are revised the list of information included in SR QU-F2 should be followed in the revision.	This is a documentation issue. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.
[redacted]					
23	QU-10	B	Open	Key assumptions and key sources of uncertainty which influence the current quantification are not addressed in a coherent manner in the documentation.	This is a documentation issue. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.
[redacted]					
24	QU-11	B	Open	The quantitative definition used for significant cutset and significant accident sequence are documented and vary from the ASME definition. The ASME definitions need to be applied or the Ameren definition needs to be justified. Significant sequence: ASME – aggregate 95% of total, individual sequence >1% Ameren – aggregate 88% of total, individual sequence >1% Significant cutset: ASME – aggregate 95% of total, individual cutset >1% Ameren – cutsets >1E-6	This is a documentation issue. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.
[redacted]					

Table 2.1.3
Callaway PRA Gap Analysis "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F/O Description	Comments / ILRT Disposition
25	LE-1	B	Open	Probability of containment isolation failure leading to LERF does not contain a term to represent undetected, residual failures in containment structural integrity. This has been estimated at 5E-3 in NUREG/CR-4550. Failure of containment isolation is derived by fault tree analysis of the containment isolation combinations on the penetration paths. There are three LERF split fractions with probabilities of 7.7E-4. If the 5E-3 was added to this, the split fraction would change, although LERF would not move significantly. Split fractions for induced SGTR and HPME were not explicitly stated in the documentation available for review.	Split fractions for SGTR and HPME <u>were</u> included in the LERF analysis. An undetected, residual failure to the containment is evaluated explicitly by the one-time ILRT extension. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.
[redacted]					
26	LE-2	B	Open	The Level 2 analysis does not include uncertainty analysis nor are there sensitivity studies identified to examine the significant contributors to LERF. As a minimum, the Uncertainty in the Level 1 sequences should be propagated and sensitivity studies developed and evaluated for the important LERF scenarios.	This is a documentation issue. Therefore, this F/O would not impact the results of the PRA evaluation for the one-time ILRT extension.

2.2 Callaway PRA WOG Peer Review

A peer review of the Callaway PRA has been performed by the Westinghouse Owner's Group (WOG) in accordance with NEI 00-02 before the Gap Analysis. The WOG Peer Review evaluated both the technical quality and adequacy of Callaway PRA and the PRA maintenance and update process. There were four "A" level Fact & Observations (F&Os) and twenty-eight "B" level F&Os were identified during the peer review process. Resolution of all F&Os from the peer review is essentially complete with a few exceptions. Table 2.2.1 provides the justification of the five F&Os identified in the Peer Review that remain opened for their impacts on the risk assessment for ILRT extension request. These open items would not impact the results of the risk assessment for the ILRT extension.

Table 2.2.1
Callaway PRA Peer Review "A/B" F&Os ILRT Disposition

Item	F/O	Level	Status	F&O Description	Comments / ILRT Disposition
1	IE-7	B	Open	Two interfacing system LOCA issues: 1. ISLOCA locations are limited to only those scenarios where containment may be bypassed. There are several lines where ISLOCAs can occur and lead to a loss of coolant inside containment. 2. The ISLOCA quantification does not correlate variables for basic events using the same failure rate. For basic events that appear together in a cut set (i.e., RHR suction valves) and have a large error factor, this can affect the results greatly.	In the Callaway PRA, an ISLOCA event is assumed to result in core damage and large early release. No mitigation is credited. An ISLOCA event is a small contributor to CDF. Thus, issue 1 would only have a potentially minor impact on CDF. Issue 2 may impact an uncertainty analysis. However, it is unlikely that an uncertainty analysis that addresses this issue via a "state-of-knowledge" correlation would preclude an application. While ISLOCA is a major contributor to LERF, an ISLOCA event is not impacted by containment isolation capability. Therefore, this F&O should not impact the results of the PRA evaluation for the one-time ILRT extension.
2	ST-1	B	Open	The ISLOCA analysis did not use current state of the art analysis to determine probability of low pressure pipe failure upon overpressure, such as the approach indicated in references such as NUREG/CR-5102 or NUREG/CR-5744.	In the Callaway PRA, an ISLOCA event is assumed to result in core damage and large early release. No mitigation is credited. While ISLOCA is a major contributor to LERF, an ISLOCA event is not impacted by containment isolation capability. Therefore, this F&O would not impact the results of the PRA evaluation for the one-time ILRT extension.
3	TH-3	B	Open	Consider preparing success criteria guidance for the PRA, to address such items as overall success criteria definition process, development of success criteria for systems, etc.	This is a documentation issue. No issues were identified with the actual success criteria utilized. Therefore, this F&O would not impact the results of the PRA evaluation for the one-time ILRT extension.
4	L2-1	A	Open	Address containment isolation failure and internal floods in the LERF calculation.	An undetected, residual failure to the containment is evaluated explicitly by the one-time ILRT extension. Likewise, Internal Flooding was addressed explicitly in the one-time ILRT extension. Therefore, this F&O would not impact the results of the PRA evaluation for the one-time ILRT extension.

Table 2.2.1 Callaway PRA Peer Review "A/B" F&Os ILRT Disposition					
Item	**F/O**	**Level**	**Status**	**F&O Description**	**Comments / ILRT Disposition**
5	L2-3	B	Open	The calculation of LERF is based on containment event tree split fractions. The process simply multiplies the split fractions together, resulting in an overall LERF split fraction for each PDS. It is not obvious how the split fractions are related back to elementary phenomena or system failures.	The Callaway process of using split fractions to partition a PDS to a LERF status is similar to the process used in NUREG/CR-6595. The split fractions are not generally subjected to change due to system failures. Any systems that were credited in accident mitigation (e.g., sprays or CHR) were explicitly modeled, not developed as split fractions. Elementary phenomena (such as DCH due to corium dispersal that is dependent on a plant's cavity design) do not usually change, and thus split fractions do not change. Containment isolation failure is not subject to split fractions. Therefore, this F&O would not impact the results of the PRA evaluation for the one-time ILRT extension.

3.0 CONCLUSION

The risk assessment for Callaway ILRT (Type A) interval extension request follows the guidelines from NEI 94-01, Revision 2 and the methodology used in EPRI Report No. 1009325, Revision 2. Capability Category I of ASME RA-Sa-2003 is used as the standard for the application since approximate values of CDF and LERF and their distribution among release categories are sufficient for use in the EPRI methodology. The Callaway PRA group has conducted a Gap Analysis and WOG Peer Review in which a number of Fact and Observations (F&Os) were identified. Tables 3 and 4 provide the justification of the F&Os that are remaining open for their impacts on the risk assessment for ILRT interval extension request. These open items are determined to not impact the results of the risk assessment for the ILRT extension. The technical adequacy of Callaway's PRA is sufficient to support the license amendment request for ILRT interval extension.

4. REFERENCES

[1] *Final Safety Evaluation for NEI Topical Report (TR) 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2 (TAC No. MC9663)*, U.S. Nuclear Regulatory Commission, Project No. 689, June 25, 2008.

[2] *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, NEI 94-01, Revision 2, August 2007.

[3] *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, EPRI, Palo Alto, CA, Report No. 1009325, Revision 2, August 2007.

[4] *Callaway PRA Gap Analysis Report*, Scientech, LLC, September 2006.

[5] *Callaway Plant Probabilistic Risk Assessment Peer Review Report*, Westinghouse Owners Group, January 2002.