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CENG
a joint venture of
Constellation Energy
eDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

October 4, 2010

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 2; Docket No. 50-318
License Amendment Request: One-Time Extension of the Containment Integrated Leakage Rate Test Interval

Pursuant to 10 CFR 50.90, Calvert Cliffs Nuclear Power Plant, LLC (Calvert Cliffs) requests an amendment to the Renewed Operating License No. DPR-69 for Calvert Cliffs Unit No. 2. The proposed amendment revises Calvert Cliffs Technical Specification 5.5.16, “Containment Leakage Rate Testing Program” to allow a one-time extension of the Type A Integrated Leakage Rate test interval for no more than five years.

The significant hazards discussion and the technical basis for this proposed amendment are provided in Attachment (1). The marked up page of the affected Technical Specification is provided in Attachment (2).

The proposed amendment is risk-informed and follows the guidance in Regulatory Guide 1.200, Revision 2. Calvert Cliffs performed a plant-specific evaluation to assess the risk impact of the proposed amendment. A copy of the risk assessment is provided in Attachment (3).

Calvert Cliffs requests approval of this proposed amendment by February 1, 2011 with an implementation period of 45 days. Approval by this time will allow Calvert Cliffs to avoid performing final preparations that would otherwise be necessary to conduct an integrated leakage rate test during the Unit 2 refueling outage that is scheduled to begin in February 2011.

A017
NRR

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, George H. Gellrich, being duly sworn, state that I am Vice President - Calvert Cliffs Nuclear Power Plant, LLC (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of St. Mary's, this 4th day of October, 2010.

JANET L. SCULLY NOTARY PUBLIC SAINT MARY'S COUNTY MD

WITNESS my Hand and Notarial Seal:

Janet L. Scully
Notary Public

My Commission Expires:

March 1, 2011
Date

GHG/KLG/bjd

Attachments: (1) Evaluation of the Proposed Change
Enclosure (1) Unit 2 Type B and Type C LLRT Results
(2) Unit 2 Type B and Type C LLRT Schedule
(2) Marked up Technical Specification Page
(3) Risk Assessment of the Proposed Amendment
(4) Statement of Calvert Cliffs Probabilistic Risk Assessment (PRA) Quality
Enclosure (1) Technical, Non-Documentation Findings

cc: D. V. Pickett, NRC
M. L. Dapas, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (1)

EVALUATION OF THE PROPOSED CHANGE

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1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend the Renewed Operating License DPR-69 for Calvert Cliffs Nuclear Power Plant (Calvert Cliffs) Unit 2. The proposed change would revise Calvert Cliffs Technical Specification 5.5.16, "Containment Leakage Rate Testing Program" by adding an exception, which would permit a one–time extension of the Containment Type A Integrated Leakage Rate Test (ILRT) interval from 10 to 15 years for Calvert Cliffs Unit 2.

2.0 DETAILED DESCRIPTION

Calvert Cliffs Technical Specification 5.5.16, "Containment Leakage Rate Testing Program" currently states, in part:

> "A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, including errata, as modified by the following exceptions:
>
> a. Nuclear Energy Institute (NEI) 94-01 – 1995, Section 9.2.3:
>
> The first Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007. ..."

The proposed change to Calvert Cliffs Technical Specification 5.5.16, "Containment Leakage Rate Testing Program" will add a similar exception for a Type A test interval extension from 10 to 15 years for Unit 2. The proposed change will revise Technical Specification 5.5.16 to state, in part:

> "A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, including errata, as modified by the following exceptions:
>
> a. Nuclear Energy Institute (NEI) 94-01 – 1995, Section 9.2.3:
>
> The first Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007. The first Unit 2 Type A test performed after the May 2, 2001 Type A test shall be performed no later than May 1, 2016. ..."

A markup of Technical Specification 5.5.16 is provided in Attachment (2).

This proposed change is requested to delay the performance of the next Unit 2 ILRT from the 2011 refueling outage to a subsequent refueling outage (no later than May 1, 2016) when it can be performed in a refueling outage that involves fewer conflicts with other planned activities and without extending the refueling outage duration.

Attachment (3) contains the plant specific risk assessment conducted to support this proposed change. This risk assessment followed the guidelines of Nuclear Regulatory Commission (NRC) Regulatory Guide 1.174 (Reference 1) and NRC Regulatory Guide 1.200, Revision 2 (Reference 2). The risk assessment concluded that the increase in risk as a result of this proposed change is small and is well within established guidelines.

3.0 TECHNICAL EVALUATION

3.1 Containment Building Description

The basic design criteria are that the integrity of the liner plate be maintained under all loading conditions and the structure shall have a low strain elastic response such that its behavior will be predictable under all design loadings.

The containment structure consists of a post-tensioned reinforced concrete cylinder and dome connected to and supported by a reinforced concrete foundation slab. The interior surface of the structure is lined with a ¼" thick welded steel plate to assure a high degree of leak tightness. The containment structure has personnel and equipment access openings as well as numerous mechanical and electrical systems that penetrate the containment structure wall through welded steel penetrations. The penetrations and access openings were designed, fabricated, inspected, and installed in accordance with the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code, Section III, Class B.

The containment structure, in conjunction with Engineering Safeguards Features, is designed to withstand the internal pressure and coincident temperature resulting from the energy released in the event of the LOCA associated with rated full power operation. The design conditions for the structure are an internal pressure of 50 psig, a coincident concrete surface temperature of 276°F and a leak rate of 0.16% by weight per day at design temperature and pressure.

3.1.1 Containment Liner

The containment liner is a ¼" thick welded steel plate that is attached to the inside face of the containment structure dome, cylindrical wall, and foundation slab. It forms a leak-tight barrier against the release of radioactive material outside the containment structure. The ¼"-thick liner plate is attached to the concrete by means of an angle grid system stitch welded to the liner plate and embedded in the concrete. The frequent anchoring is designed to prevent significant distortion of the liner plate during accident conditions and to insure that the liner maintains its leak-tight integrity. The liner plate is protected from corrosion on the inside with 3 mils of inorganic zinc primer topped with 6 mils of an organic epoxy up to Elevation 75'0", and 3 mils of an inorganic topcoat above that elevation. There is no paint on the side that comes in contact with the concrete.

A finished concrete floor covers the portion of the liner on the containment foundation slab. A leak chase system allows the containment liner welds located under the concrete floor to be leak tested during the ILRT of the containment.

3.1.2 Electrical Penetrations

Two types of electrical penetration assemblies are used - canister and unitized header. All electrical penetration assemblies were fabricated and tested in accordance with the ASME, B&PV Code, Section III, Nuclear Vessel Code. The canister-type is inserted in a nozzle of suitable diameter integral with the containment structure and field welded on the inside end. The unitized header-type is welded to the nozzle on the outside end. All penetration assemblies are provided with a means to pressurize for monitoring of leakage. Any abnormal depressurization of an assembly is annunciated both locally and in the Control Room.

3.1.3 Piping Penetrations

Single barrier piping penetrations are provided for all piping passing through the containment walls. The closure of the pipe to the liner plate is accomplished with a pipe cap welded to the pipe and to the liner plate reinforcement. In the case of piping that carries hot fluid, the pipe is insulated and cooling is provided to reduce the concrete temperature to 150°F. The anchorage of penetration closure connecting pipes to the containment wall were designed as Seismic Category I structures to resist all forces and moments caused by a postulated pipe rupture. The design conditions include the maximum pipe reactions and pipe rupture forces.

The penetration assembly, consisting of pipe cap and the assembly welds and welds to the liner plate, utilizes full penetration welds. The assembly is anchored into the wall concrete and designed to accommodate all forces and moments due to pipe rupture and thermal expansion.

3.1.4 Containment Penetration Bellows Assemblies

Expansion bellows are not utilized in the design of the mechanical penetrations at Calvert Cliffs. There are bellows used on the fuel transfer tube penetration to accommodate relative movement between the refueling canal liner and the containment building penetration. However, those bellows do not form part of the containment building vessel or pressure boundary. They are unaffected by this proposed amendment.

3.1.5 Refueling Tube Penetration

A refueling tube penetration is provided for fuel movement between the refueling pool in the containment structure and the spent fuel pool in the Auxiliary Building. The penetration consists of a 36" stainless steel pipe installed inside a 42" pipe sleeve. The inner pipe acts as the refueling tube and is fitted with a gate valve in the spent fuel pool and an encapsulating pipe sleeve, which is welded to the refueling pool liner and sealed off from the Containment with a testable double O-ring blind flange in the refueling pool. This arrangement prevents leakage through the refueling tube in the event of a LOCA. The 42" pipe sleeve is welded to the containment liner.

Bellows expansion joints are provided on the transfer tube to compensate for any differential movement between the tube and the building structures. The bellows do not form any part of the containment boundary so they are unaffected by this proposed change.

3.1.6 Moisture Barrier

A layer of compressible material covers both sides of the containment liner on the containment wall where the finished concrete floor joins the wall. This cork layer, covered with a waterproof seal, serves as an expansion joint to accommodate any relative movement between the containment wall, floor, and liner.

3.1.7 Containment Tendons

The containment post-tensioning system consists of:

- Three groups of 68 dome tendons oriented at 60° to each other for a total of 204 tendons anchored at the vertical face of the dome ring girder.
- Two hundred four vertical tendons anchored at the top surface of the ring girder and at the bottom of the base slab.
- Six groups of 78 hoop tendons, each enclosing 120° of arc, for a total of 468 tendons anchored at the 6 vertical buttresses.

Each tendon consists of approximately 90 ¼" diameter wires with button-headed BBRV-type anchorages. The tendons are housed in spiral wrapped, corrugated, thin-wall, carbon steel sheathing.

After fabrication, each tendon was shop dipped in a petroleum corrosion protection material. After installation, the tendon sheathing was filled with a corrosion preventive grease. The ends of all tendons were covered with pressure-tight, grease filled caps for corrosion protection. All the vertical tendons for each unit have received new corrosion preventive grease between 1997 and the end of 2002. In addition some original vertical tendons for each unit were restressed or replaced with new tendons between 2001 and 2002.

In the concept of a post-tensioned containment structure, the internal pressure load is balanced by the application of an opposing external force on the structure. Sufficient post-tensioning was used on the cylinder and dome to more than balance the internal pressure so that a margin of external pressure exists beyond that required to resist the design pressure. Nominal, bonded reinforcing steel was also provided to distribute strains due to shrinkage and temperature. Additional bonded reinforcing steel was used at penetrations and discontinuities to resist local moments and shears.

The internal pressure loads on the foundation slab are resisted by both the external bearing pressure due to dead load and the strength of the reinforced concrete slab. Thus, post-tensioning was not required to exert an external pressure for this portion of the structure.

3.2 Justification for the Technical Specification Change

The performance-based ILRT requirements of Option B of 10 CFR Part 50, Appendix J, provide an alternative to the three tests per ten-year frequency specified by the prescriptive requirements of Option A of 10 CFR Part 50, Appendix J. As documented in Regulatory Guide 1.163 (Reference 4), the NRC has endorsed NEI 94-01 (Reference 5) as providing acceptable methods for complying with the requirements of Option B of 10 CFR Part 50, Appendix J. Nuclear Energy Institute 94-01 (Reference 5) specifies an ILRT frequency of one test per ten years provided certain performance criteria are met. The basis for the one test per ten-year frequency is described in Section 11.0 of NEI 94-01 (Reference 5), which references NUREG-1493 (Reference 6), as providing the technical basis to support rulemaking that established Option B. That 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.

The Electric Power Research Institute (EPRI) undertook a similar study, the results of which are documented in EPRI report Topical Report (TR)-104285, (Reference 7). The EPRI study determined a reduction in the frequency of ILRTs from three tests per ten years to one test per ten years would result in an incremental risk contribution of 0.035%. This value is comparable to the range of risk increases (0.02% to 0.14%) presented in NUREG-1493 for the same frequency reduction. Additionally, NUREG-1493 described the increase in risk resulting from an even lower frequency, one test per 20 years, as "imperceptible."

3.2.1 Current Calvert Cliffs ILRT Requirements

Title 10 CFR Part 50, Appendix J requires periodic tests to assure that leakage through the primary reactor containment and systems and components penetrating primary containment does not exceed the allowable leakage rate values as specified in the Technical Specifications. Appendix J requires three types of tests: 1) Type A tests, intended to measure the primary containment overall integrated leakage rate; 2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries for primary containment penetrations other than valves; and 3) Type C tests, intended to measure containment isolation valve leakage rates. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the Containment not covered by Type B and C testing.

Title 10 CFR Part 50, Appendix J was revised, effective October 26, 1995, to allow licenses to choose containment leakage testing under either Option A, "Prescriptive Requirements,", or Option B, "Performance Based Requirements." On March 13, 1996 the NRC approved License Amendment No. 189 for Calvert Cliffs Unit 2 authorizing the implementation of 10 CFR Part 50, Appendix J, Option B for Type A tests. Current Technical Specification 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 Part 50 Appendix J, Option B, as modified by approved exemptions. The program is required to be in accordance with the guidelines contained in Regulatory Guide 1.163 (Reference 4). Regulatory Guide 1.163 endorses, with certain exceptions, NEI 94-01 (Reference 5) as an acceptable method for complying with the provisions of Appendix J, Option B.

Regulatory Guide 1.163, Section C.1 states that licensees intending to comply with 10 CFR Part 50, Appendix J, Option B, should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 (Reference 5) rather than using test intervals specified in ANSI/ANS-56.8-1994. Nuclear Energy Institute 94-01 (Reference 5), Section 11.0 refers to Section 9, which states that Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per ten 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 was less than 1.0 L_a (where L_a is the maximum allowable leakage rate at design pressure). Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

Adoption of the Option B performance based containment leakage rate testing program altered the frequency of measuring primary containment leakage in Types A, B, and C tests but did not alter the basic method by which Appendix J leakage testing is performed. The test frequency is based on an evaluation of the "as found" leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will not be exceeded. The allowed frequency for Type A testing as documented in NEI 94-01 (Reference 5), is based, in part, upon a generic evaluation documented in NUREG-1493 (Reference 6). The evaluation documented in NUREG-1493 included a study of the dependence or reactor accident risks on containment leak tightness for differing types of containment types, including a post tensioned, shallow domed concrete containment similar to Calvert Cliffs containment structures. NUREG-1493 concluded in Section 10.1.2 that reducing the frequency of Type A tests (ILRT) from the original three tests per ten years to one test per twenty 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 Types 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, NUREG-1493 concluded that increasing the interval between ILRTs is possible with minimal impact on public risk.

Title 10 CFR Part 50, Appendix J, Option B, Section V.B, allows exceptions to the guidelines of Regulatory Guide 1.163. That section states; "The regulatory guide or other implementation document used by a licensee or applicant for an operating license under this part or a combined license under Part 52 of this chapter to develop a performance-based leakage-testing program must be included, by general reference, in the plant Technical Specifications. The submittal for Technical Specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide." Since exceptions meeting the stated requirements are permitted, this license amendment request does not require an exemption from 10 CFR Part 50, Appendix J, Option B.

As noted previously, Calvert Cliffs Technical Specification 5.5.16 requires Type A, B, and C testing in accordance with Regulatory Guide 1.163, which endorses the methodology for complying with Option B. The performance leakage rates are calculated in accordance with NEI 94-01 (Reference 5), Section 9.1.1. The performance leakage rate includes the Type A Upper Confidence Limit at 95% plus the as-left minimum pathway leakage rate for all Type B and C pathways not in service, isolated, or not lined up in their test position.

Unit 2 Type A ILRT History

Test Date	Leakage Rate (Containment air weight%/day)	As Left Type C Minimum Path Contribution weight%/day
03/14/1976	0.019 %/day	0.0001 %/day
11/15/1979	0.052 %/day	0.00084 %/day
12/22/1982	0.025 %/day	0.0015 %/day
11/24/1985	0.185 %/day	0.081 %/day

The results of the last two Type A ILRT for Calvert Cliffs Unit 2 are listed below:

Test Date	Leakage Rate (Containment air weight%/day)	As Left Type C Minimum Path Contribution weight%/day
01/16/1991	0.061 %/day	0.001 %/day
05/02/2001	0.0738 %/day	0.0014 %/day

Both results of the last two Type A ILRTs are less than the maximum allowable containment leakage rate of 0.16 %/day at the test pressure of 50 psig. As a result, since both tests were successful, the current ILRT interval frequency for Calvert Cliffs Unit 2 is ten years.

3.2.2 Steam Generator Replacement

During the 2003 refueling outage Calvert Cliffs replaced Unit 2s steam generators. The steam generator replacement affected only the closed piping inside Containment as the new steam generator assemblies and the old steam generator assemblies transited through the containment equipment hatch. The containment structure and the containment liner were not affected. However, the steam generator shell and the inside-containment portions of the main steam, feedwater, steam generator blowdown, and auxiliary feedwater lines were considered an extension of the primary reactor containment. As a result, Calvert Cliffs submitted a request (Reference 14) for an alternative to the technical specification requiring the performance of an ILRT following the replacement of Unit 2s steam generators. In this request Calvert Cliffs indicated that the required ASME testing for the affected piping systems and steam generator shell provided an acceptable alternative method which would test only the modified portions of the containment barrier instead of the more comprehensive Type A testing which would be performed on the entire containment barrier. Following review of this proposed alternative, the NRC issued a safety evaluation (Reference 15) that approved this change based on the determination that the required ASME examinations and testing requirements were more stringent than the Type A testing.

3.3 Risk Based Assessment

As part of this proposed amendment change an assessment was performed of the risk impact of a one time extension of Calvert Cliffs Unit 2 Type A ILRT frequency to 15 years. Although Calvert Cliffs Unit 2 current ILRT frequency is one test per ten 10 years based on acceptable, past ILRT performance, the risk assessment evaluated the risk increase from the original three tests per ten years to one test per fifteen years frequency. Even using this more conservative approach, the proposed extension was found to have a very small increase in risk (significantly less than 1% of the total integrated plant risk).

The risk assessment followed the guidelines from NEI 94-01 (Reference 16), the methodology used in EPRI TR-1009325 (Reference 17), and the NRC regulatory guidance on the use of PRA findings and risk insights as outlined in Regulatory Guide 1.174 (Reference 1). Although this methodology generally produces more conservative results than do the earlier methodologies, they build upon the work of the earlier studies, and much of the analyses developed from application of the EPRI TR-104285 methodology (Reference 7) remains applicable for use in these more recent studies. Comparison sensitivity studies were also performed with alternate approaches. The risk assessment results for Calvert Cliffs are consistent with those of previous studies supporting other plants' ILRT extension requests. The following are the conclusions from the completed risk assessment associated with extending this Type A ILRT test:

- There is no change in the at-power core damage frequency (CDF) associated with this ILRT test interval extension from 10 to 15 years. Therefore, this is within the Regulatory Guide 1.174 acceptance guidelines.
- Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10-6/yr and increases in large early release frequency (LERF) below 10-7/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from three test per ten years to one test per fifteen years is less than 1E-07/yr. Therefore, increasing this ILRT interval from 10 to 15 years is considered to result in a very small change to the Calvert Cliffs risk profile based on the Regulatory Guide 1.174 definition.
- The proposed change in the Type A test frequency (from three tests per ten years to one test per fifteen years) increases the total integrated plant risk by less than 1% for Calvert Cliffs Unit 2. Therefore, the risk impact of this change, when compared to other severe accident risks, is negligible. The change in conditional Containment failure probability of approximately 1% (based on conservative methodology) is also judged to be insignificant and reflects sufficient defense-in-depth.

The above results demonstrate that the increases in total integrated plant risk and LERF resulting from the proposed amendment are within established Regulatory Guide 1.174 guidelines and that defense-in-depth principle would be maintained. The complete Calvert Cliffs risk assessment is provided in Attachment (3).

3.4 Comparison of Calvert Cliffs PRA Methodology to Regulatory Guide 1.200

The Calvert Cliffs internal events PRA model was peer reviewed in June 2010. All draft findings which had significant impact on this analysis have been addressed. This assessment is provided as Attachment (4). The ILRT application was determined to be a Category II application of the Regulatory Guide 1.200 criteria, Revision 2. 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. Attachment (4) includes discussion of all draft findings from the industry peer review along with the assessment and evaluation of the finding that shows that they have either been addressed or have no material impact on the ILRT interval extension request.

3.5 Non-Risk Based Assessment

Consistent with the defense-in-depth philosophy discussed in Regulatory Guide 1.174, Calvert Cliffs has assessed other non-risk based considerations relevant to the proposed amendment. Calvert Cliffs has multiple inspections and testing programs that ensure the containment structure remains capable of meeting its design functions and that are designed to identify any degrading conditions that might affect that capability. The implementation of these programs are not affected by the proposed change to the Type A test frequency. These programs are discussed below.

3.5.1 Containment Leakage Rate Testing Program - Type B and Type C Testing Program

Calvert Cliffs Types B and C testing program requires testing of electrical penetrations, airlocks, hatches flanges, and containment isolation valves in accordance with 10 CFR Part 50, Appendix J, Option B, and Regulatory Guide 1.163. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Types B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. Per Technical Specification 5.5.16, the allowable maximum pathway total Types B and C leakage is 0.6 L_a. L_a equals approximately 276,800 sccm.

Tables 3.5.1 and 3.5.2 provide Local Leakage Rate Test (LLRT) data trend summaries for Calvert Cliffs Unit 2 since the performance of the 2001 ILRT. This summary shows that there has been no As-Found (AF) failure that resulted in exceeding the Technical Specification 5.5.16 limit of 0.6 L_a (166,080 sccm) and demonstrates a history of successful tests.

Table 3.5.1, Unit 2 Types B and C LLRT Combined As-Found/As-Left Trend Summary

RFO	2003	2005	2007	2009
AF MAX PATH (sccm)	12051.84	15759.7	14943.1	26859.7
Fraction of L_a	0.035	0.046	0.043	0.078
AF MIN PATH (sccm)	10535.95	14380.4	10689.8	14570.3
Fraction of L_a	0.030	0.042	0.031	0.042
AL MAX PATH (sccm)	12347.1	3848.9	13936.6	11969.8
Fraction of L_a	0.036	0.011	0.040	0.035
AL MIN PATH (sccm)	11091.6	2784.9	9070.2	7028.9
Fraction of L_a	0.032	0.008	0.026	0.020

Table 3.5-2 identifies the number of Types B and C LLRTs which were found to exceed their Admin limits (assigned limit that is less than the Technical Specification limit) which results in reducing the length of time between subsequent LLRTs for that component.

Table 3.5.2, Unit 2 As-Found LLRTs Exceeding Admin Limit Summary

RFO	2003	2005	2007	2009
Number AF LLRTs Exceeding Admin Limit	2 Type C 0 Type B	3 Type C 0 Type B	5 Type C 0 Type B	8 Type C 0 Type B

The As-Found and As-Left results of all Types B and C tests performed (including the 2001 ILRT) are shown in Enclosure (1).

As outlined in NEI 94-01 (Reference 5), when eligible based on good performance, the test interval for Type C tested components may be increased up to a maximum of 60 months. One measure of the performance of Type C tested components is the percentage of eligible components that are on extended intervals. For Calvert Cliffs Unit 2, out of a possible 59 Type C performance-based components, 81.3% (48) are currently on a 60-month extended test interval. Additionally, out of a possible 67 Type B performance based components, 98.5% (66) are currently on a 120-month extended test interval. As illustrated above, the combined Types B and C leak rate has been maintained significantly below the 0.6 L_a acceptance criterion. These leak rate test results demonstrate "generally good performance" of the Types B and C tested components at Calvert Cliffs Unit 2.

As previously noted, Types B and C testing evaluates all but a small portion of the potential containment leakage pathways. This proposed amendment does not affect the scope, performance, or scheduling of Types B or C tests. Types B or C tests will continue to be performed at their scheduled frequency throughout the extension period. The proposed scheduling of Types B and C testing is shown in Enclosure (2). This helps to provide a continued high degree of assurance that primary containment integrity is maintained throughout the extension period.

3.5.2 Safety-Related and Controlled Protective Coatings Inspection Program

The requirements of 10 CFR Part 50, Appendix B are implemented through specification of appropriate technical and quality requirements for the Service Level 1 coatings program which includes ongoing maintenance activities. Calvert Cliffs has implemented controls for the procurement, application, and maintenance of Service Level 1 protective coatings used inside the Containment in a manner that is consistent with the licensing basis and regulatory requirements applicable to Calvert Cliffs.

Calvert Cliffs conducts condition assessments of Service Level 1 coatings inside Containment as part of the safety-related and controlled protective coatings program. Inspections of coatings systems are scheduled every outage on a pre-established basis to verify containment liner coating thickness and condition.

3.5.3 Containment Inservice Inspection Program

The purpose of the Calvert Cliffs containment inservice inspection (ISI) program is to periodically perform destructive and nondestructive examination of ASME Class MC and CC components in order to identify the presence of any service-related degradation. The containment ISI program is established in accordance with 10 CFR 50.55a. This program has been developed to comply with ASME Section XI 2004 Edition, except where specific written alternatives from Code requirements have been requested by Calvert Cliffs and granted by the NRC.

The program defines the Class MC and CC components and the Code-required examinations for each ASME Section XI examination category, and the augmented inspection scope, as applicable.

The components subject to the requirements of this containment ISI program are those which make up the containment structure, its leak tight barrier (including integral attachments) and those which contribute to its structural integrity, specifically, Class MC pressure-retaining components, and their integral attachments and Class CC post tensioned concrete containments.

The administrative procedures and inspection schedule described in the containment ISI program, combined with applicable Calvert Cliffs and approved vendor procedures, constitute the containment ISI portion of the ten-year ISI program.

IWE (Class MC) Inspection Interval and Periods

The second ten-year containment ISI interval for both Units for the performance of containment ISI (IWE) complies with IWE-2412 Inspection Program B and began on September 9, 2009 and will end on September 9, 2018. This interval is shortened as a result of extending the first ten year containment ISI interval by one year. The interval is then further divided into three periods which are as follows:

1st Period: September 9, 2009 through September 9, 2011 (2 years)

2nd Period: September 9, 2011 through September 9, 2015 (4 years)

3rd Period: September 9, 2015 through September 9, 2018 (3 years)

IWL (Class CC) Inspection Periods (Concrete)

The second ten-year containment interval for the performance of containment ISI (IWL) for both Units complies with IWL-2400 and is effective for IWL inspections conducted between September 9, 2009 and September 9, 2018.

Concrete examinations shall be conducted every five years (+/- one year) as described in IWL-2410 (a) and (c). For the purposes of the containment ISI program, an IWL inspection period is five years, with two periods per inspection interval.

Concrete surface areas affected by a repair/replacement activity shall be examined at one year (+/- three months) following completion of repair/replacement activity. If plant operating conditions are such that examination of portions of the concrete cannot be completed within this time interval, examination of those portions may be deferred until the next regularly-scheduled plant outage.

IWL (Class CC) Inspection Periods (Tendons)

For multiple-unit plant sites, such as Calvert Cliffs, the tendon examination frequency may be extended to ten years per unit provided the containment structures utilize the same pre-stressing system, are essentially identical in design, had their original structural integrity test performed within two years of one another, and experience similar environmental exposure. The examinations required by IWL-2500 for unbonded post-tensioning systems can then alternate between the two units every five years, as allowed by IWL-2421 (sites with multiple units).

Going forward for Calvert Cliffs Unit 2, the following two ASME required tests are to be performed once every ten years.

- Tendon force and elongation measurements (tendon lift-off test)
- Tendon wire and strand sample examination and testing (wire removal tensile test)

These tests are next scheduled to be performed no later than 2013.

The following three ASME required tests are to be performed once every five years.

- Examination of tendon anchorage areas (visual examination)
- Sampling and analysis of corrosion protection medium (grease analysis)
- Free water collection and analysis (free water analysis)

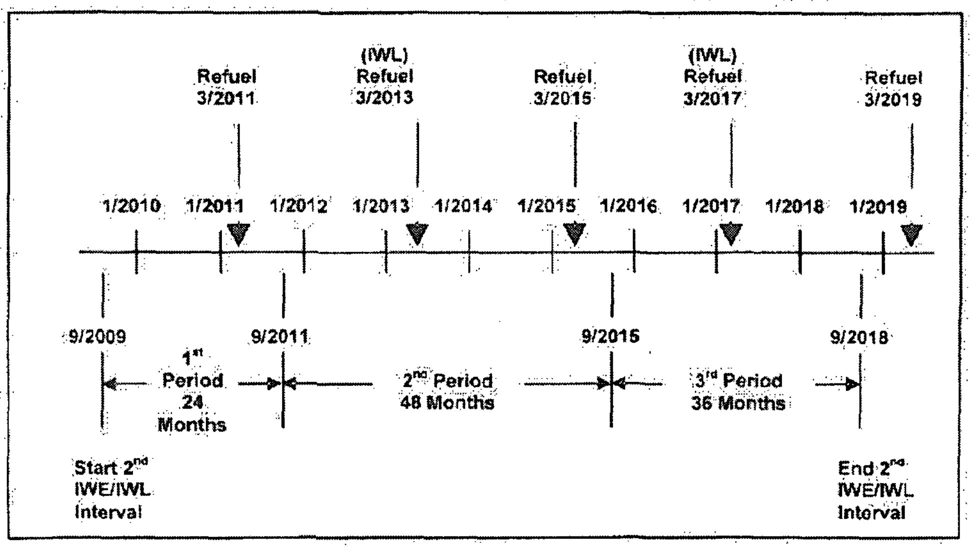
These tests are also next scheduled to be performed no later than 2013. Figure 3.5.1 shows the timeline for the IWE/IWL inspections during the second ten-year containment ISI interval.

Figure 3.5.1

CCNPP Unit 2 IWE/IWL Examination Periods

Period	Date	Tolerance
35 year	9/9/2012	+/- 1 Year
40 Year	9/9/2017	+/- 1 Year

CCNPP Unit 2 IWE/IWL Interval, Periods, and Outages



Adoption of Code Cases

All Code Cases adopted for ASME Section XI activities for use during the second ten-year containment ISI interval are listed below. The use of Code Cases is in accordance with ASME Section XI, IWA-2440, 10 CFR 50.55a, and Regulatory Guide 1.147 (Reference 18). As permitted by ASME Section XI and Regulatory Guide 1.147 or 10 CFR 50.55a, ASME Section XI Code Cases may be adopted and used as described below.

Code Cases Adopted from Regulatory Guide 1.147

N-532-4 Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission

N-624 Successive Inspections

N-686 Alternative Requirements for Visual Examinations, VT-1, VT-2, and VT-3

N-739 Alternative Qualification Requirements for Personnel Performing Class CC Concrete and Post-Tensioning System Visual Examinations. This fulfills NRC concerns stated in 10 CFR 50.55a(b)(2)(ix)(F) regarding "owner-defined" personnel qualifications

Relief Requests

Calvert Cliffs has submitted no requests for alternatives or requests for relief applicable to the requirements for Class MC and CC components.

Component Exemptions, IWE and IWL

The basis for the selection of components at Calvert Cliffs which are determined to be within the scope of the required examinations was done in accordance with the requirements of IWE-1220 and IWL-1220 respectively.

Calvert Cliffs does have areas that are considered inaccessible which are therefore exempt from inspection and are described below:

IWE – The containment liner covering the containment foundation slab is inaccessible. This area is covered with the finished concrete floor and moisture barrier and accounts for approximately 15% of the containment liner surface area.

IWL – Portions of the concrete surface that are covered by the liner, foundation material or backfill, or are otherwise obstructed by adjacent structures, components, parts or appurtenances are inaccessible. The entire inside concrete surface of the Calvert Cliffs containment buildings area covered in steel which makes them inaccessible for examination.

Examination Methods & Personnel Qualifications

The examination methods used to perform Code examinations for the nonexempt Class MC and CC components are in accordance with 10 CFR 50.55a requirements and the applicable ASME Codes.

Personnel performing IWE examinations shall be qualified in accordance with Constellation Energy's written practice, or approved vendor written practice for certification and qualification of NDE personnel.

Personnel performing IWL examinations shall be qualified in accordance with written procedures prepared as required by IWL-2300, as modified by applicable Code Cases.

3.6 Unit 2 Operating Experience

During the conduct of the various examinations and tests conducted in support of the Containment related programs previously mentioned, issues that do not meet established criteria or that provide indication of degradation, are identified, placed into the site's corrective action program, and corrective actions are planned and performed.

In addition, Calvert Cliffs and our corporate organization, Constellation Energy Nuclear Group, actively participates in various nuclear utility owners groups, ASME code committees, and with NEI to maintain cognizance of ongoing developments within the nuclear industry. Industry operating experience is also continuously reviewed to determine its potential applicability to Calvert Cliffs. As a result of these reviews, adjustments to inspection plans may be made and availability of new, commercially available technologies are explored.

For Calvert Cliffs Unit 2 Containment there are three issues of degradation that have been identified that have either been corrected or are in the process of having the necessary corrective actions completed during upcoming outages. The three areas of note involve:

Moisture barrier seal degradation/Liner corrosion

Containment concrete surface degradation

Vertical tendon corrosion

Each of these areas is discussed in detail in Sections 3.7 through 3.9, respectively.

3.7 Containment Liner and Moisture Barrier Seal

3.7.1 Inspections

Inspections on the containment liner are conducted in accordance with Examination Category E-A of the ASME Code Section XI, 2004 Edition. These inspections are performed such that 100% of the accessible portion of the liner is inspected during each inspection period. As previously mentioned the portion of the liner that covers the containment foundation slab is considered inaccessible. Since this inaccessible area cannot be inspected, Calvert Cliffs must therefore evaluate its acceptability whenever conditions exist in the accessible areas that could indicate, the presence of or result in, degradation to the inaccessible area.

The moisture barrier seal is examined so that 100% of the seal is visually examined during each inspection period.

3.7.2 Inspection Results

In 1994 Calvert Cliffs discovered significant age related degradation of Unit 1s moisture barrier seal. As part of the corrective actions, a decision was made to subsequently replace Unit 2s moisture barrier seal. In 1999 during the replacement of Unit 2s moisture barrier seal, areas of pitting and general corrosion were discovered on the Unit 2s metal containment liner that exceeded 10% of the nominal wall thickness of ¼". The liner area of concern was the wall to floor transition under the moisture barrier seal, between the wearing floor slab and the containment liner wall. The worst area of general corrosion was visually determined to be located between vertical leak channels #1 and #2. To conduct a complete assessment of this area, all compressible material was removed from between the liner wall and the floor slab over an area 32" wide by 12" deep (down to the first horizontal leak channel where the ¼" thick liner transitions to a ½" thickness). This area was then ultrasonically tested and the thinnest wall thickness measured was 0.20".

The deepest pitting was measured in the vicinity of vertical leak channel # 30 located 1.38" below the wearing floor grade. Actual wall thickness adjacent to the worst pit was measured as 0.29" by ultrasonic testing. Worst pit depth was measured as 0.19" using a pit gage. Remaining wall thickness beneath the pit was determined to be 0.10" by computation.

An evaluation was subsequently performed which determined that if the degradation was not stopped, but instead continued at its current rate, the pitted areas would degrade further and pose a concern in the future. While it was impossible to determine when the pitting began, it was reasonable to assume that the degradation would be stopped or significantly slowed by the replacement of the moisture barrier seal.

As part of the evaluation, consideration was given as to whether additional areas needed to be examined. A determination was made that no additional examinations of other areas were necessary.

The evaluation also concluded that it was acceptable, in accordance with ASME Section XI, 1992 Addendum, Subsection IWE, Article IWE-3122.4, to return the liner to service without repair of the degraded area since the area of degradation is non-structural in nature and has no effect on the structural integrity of Containment.

The replacement of the moisture barrier seal involves use of a new seal material (high density silicone elastomer (HDSE)) that provides an effective seal against water, smoke, gas, and pressure. Along with the installation of the new HDSE sealant, a modification to the design of the seal was done. The original base sealant was applied to a shallow depth at the top of the compressible material in the joints and made flush with the nominal base slab. The new HDSE sealant was installed in such a manner to form a small curb above the joint which would shed water in addition to providing a seal. Also, to improve the seal, the HDSE was placed a minimum of 3" into the joint by removing some of the compressible material. A polyethylene backer rod was then placed in the joint between the HDSE and the compressible material to separate them.

Replacement of the Unit 2 moisture barrier seal is almost complete. Only a small area around the pedestal/finish slab joint and the buttress/finish slab joint remains. These repairs are scheduled for completion by the end of the Unit 2 2013 refueling outage as there are no indications of water intrusion or general corrosion occurring in those areas.

The most recent inspections of the containment liner and moisture barrier seal indicate that the replacement of the moisture barrier seal has arrested the corrosion and pitting throughout the affected area and has prevented any new areas of corrosion and pitting from occurring. As a result the liner continues to be acceptable to perform its safety function (i.e., act as a leak tight membrane).

3.8 Containment Concrete

3.8.1 Concrete Inspections

The reinforced concrete portions of Containment are inspected in accordance with Examination Category L-A of the ASME Code Section XI, 2004 Edition. The concrete containment structure is divided into 115 areas on Unit 2. Calvert Cliffs conducts a 100% visual examination of each unit every five years.

3.8.2 Inspection Results

The most recent containment concrete inspections were conducted during 2005 and 2007. In these inspections examiners identified new grease leaks, efflorescence, and other stains. All these items were entered into the corrective action program for resolution.

The examiners also identified two issues on the Unit 2 Containment that had been identified in a previous inspection (2001) but had not been fully addressed. These items are:

- Containment structure dome area is suffering from the effects of weathering due to freeze –thaw cycles. This issue, if not addressed, would eventually pose a threat to containment integrity as water soaking into the concrete would attack the reinforcing steel. The occurrence of freeze-thaw cycles accelerates this process by breaking up the concrete surface. The proposed corrective action for this issue is to remove any loosened concrete, clean stains from around areas of major grease leaks, and apply a sealer to minimize moisture penetration. Completion of these actions for Unit 2 Containment is scheduled for December 15, 2012.

- Concrete was found to be delaminating around the sloped surface above the equipment hatch. Delamination opens the surface to water entry and could cause pieces of concrete to fall off. The proposed corrective action for this issue involves the removal of loose concrete and the application of an epoxy bonding compound to which low slump 5000 psi concrete will be applied. Completion of these actions for Unit 2 Containment is scheduled for December 15, 2012.

An evaluation of these two issues determined the concrete in those areas is still capable of maintaining its structural integrity in the event of a design basis LOCA and that it will continue to perform this function beyond the completion date for the repairs.

3.9 Containment Tendons

3.9.1 Containment Tendon Inspections

The containment tendons are inspected in accordance with Examination Category L-B of the ASME Code Section XI, 2004 Edition. Figure 3.9.1 below shows the tendon population distribution for Unit 2.

Figure 3.9.1, Calvert Cliffs Unit 2 Tendon Population Distribution

	Dome	Hoop	Vertical (Original)	Vertical (Replaced / Restressed)	Total
Unit 2	204	468	125	79	876

The ASME required tendon lift-off test is conducted on a minimum of 25 tendons once every ten years. Per the ASME Code, a sample of each of the four types of active tendons must be examined. The sample selections by type are as follows:

- Dome: 5 (1 Common and 4 Random)
- Hoop: 10 (1 Common and 9 Random)
- Vertical (Original-Undisturbed): 6 (1 Common and 5 Random)
- Vertical (Replaced or Restressed): 4 (0 Common, 4 Random)

A common tendon is a tendon that is tested each time the test is performed. A random tendon is a tendon selected for this test and is not selected again in subsequent tests. The replaced or restressed tendons do not have a common tendon because these tendons were recently replaced as a result of the tendon issues discussed in Section 3.9.2 below.

The ASME required wire removal tensile test is conducted once every ten years on a minimum of one tendon from each of the four tendon types.

The ASME required visual examinations of the tendon anchorage area, the free water analysis and the analysis of the corrosion protection medium (grease analysis) are performed on a minimum of 25 tendons every five years. Each of these three tests is performed on tendons that are selected so as to have the same distribution between the four tendon types as for the tendon lift-off tests.

3.9.2 Inspection Results

In 1997, during the performance of the 20-year (time from first tendon inspection) tendon surveillance on Unit 1, conditions that did not meet the acceptance standards were found on some of the

Containment tendons. Conditions that did not meet the acceptance standards were found in all three containment tendon populations, i.e., hoop, dome, and vertical tendons. The abnormal conditions found on the hoop and dome tendons were considered minor enough that the acceptability of the concrete containment was not affected. However the conditions found on the vertical tendon population were more significant. Several of the vertical tendons selected for the surveillance were found to contain broken and corroded wires at their top ends, just below the stressing washer. The discovery of broken wires in these tendons initiated an expansion of the Unit 1 vertical tendon inspection scope to perform visual inspections and lift-off testing on all Unit 1 vertical tendons. Subsequently, broken and corroded wires were found throughout the Unit 1 vertical tendon population at the top ends of the tendons. Following completion of the Unit 1 surveillance, the 20-year surveillance of the Unit 2 tendons was conducted. Although Unit 2 was only required to perform visual inspections, it was decided to also perform lift-off testing of all the vertical tendons in order to facilitate inspection of the tendon wires in the region of concern [below the upper (top) stressing washer]. Abnormal conditions very similar to Unit 1 were found on the Unit 2 vertical tendons. A non-conformance report was written for every abnormally degraded condition that did not meet the acceptance criteria.

3.9.3 Corrective Actions to Address Vertical Tendon Corrosion

As a result of the corrosion and broken wires discovered on some vertical tendons during the 1997 surveillance on the Unit 1 and 2 Containments, an evaluation was conducted. The evaluation concluded that the tendon wire failures and corrosion problems resulted from a combination of water and moist air intrusion into the vertical tendon end caps (grease cans), and inadequate initial grease coverage of wires in the area just under the top stressing washer.

To address the issues identified in the evaluation, short-term and long-term corrective actions were taken. The short-term actions included spraying hot grease under the top stressing washer, reorienting the stressing shims so as to leave a gap between the shims to allow a vent path to help eliminate voids, re-greasing non-corroded vertical tendons, and resealing around the original tendon can all-thread penetrations with caulking. Additional inspections were performed in 1999 and 2000 to verify the assumptions that were considered in the evaluation and to provide additional data to help develop the long-term corrective action plan.

The goal of the long-term corrective action plan was to ensure that the Containments meet their design basis requirements until plant end-of-life. As one part of the long-term corrective action plan, all the vertical tendons were re-greased using a new corrosion inhibiting grease (Visconorust 2090-P4). The non-corroded vertical tendons were re-greased in 2000, and the tendons with less severe corrosion that were not replaced were re-greased during 2001. The remaining vertical tendon population (46 tendons per Unit) was replaced in 2001 and 2002, and had new grease put in place at that time. At the end of these corrective actions, all of the vertical tendons had a redesigned pressure-tight, grease-filled cap installed at the upper-bearing plate to prevent water intrusion. The bottom grease cap for every vertical tendon was also replaced with a new redesigned pressure-tight grease cap. The redesigned grease cap has a flange that is attached by studs and nuts to the tendon bearing plates by utilizing existing taps in the plates.

3.9.4 Enhanced Vertical Tendon Inspections

To further confirm the effectiveness of the short- and long-term corrective actions, an enhanced inspection program was initiated that consisted of a two tiered approach. The first tier involved the performance of the required, ASME Section XI Code inspections at their normal periodicity. The second tier involved enhanced visual inspections of a selected sample size of vertical tendons that would be in addition to tendons inspected as part of the ASME required inspection. The visual inspections included inspection of the anchorhead and buttonhead region to determine if any wire breaks have occurred in the area under the vertical tendon top-stressing washers. The first enhanced inspections were performed in 2005 and the second enhanced inspections were conducted in 2007. No new issues were identified as a result of these inspections.

Based on the satisfactory performance of the enhanced inspections, an assessment was conducted which determined that continuance of the enhanced inspections was not necessary. The assessment determined that the Code required inspections are sufficient to adequately determine whether tendon performance remains acceptable.

3.9.5 Latest ASME Code Inspection Results

The latest ASME Code tendon surveillance tests were conducted in 2008. For Calvert Cliffs Unit 2, as required, only the tendon anchorage area visual examination, free water analysis, and the corrosion protection medium (grease analysis) were conducted.

The evaluation of the in-service inspection results for the 30^{th} year (2008) containment IWL inspection of the Calvert Cliffs Unit 2 containment structure concluded that the containment structure has experienced no abnormal degradation of the post-tensioning system. The containment post-tensioning systems are performing in accordance with the design requirements and are expected to continue to do so for the life of the unit.

3.10 Other Containment Inspections and Tests

In addition to the inspection requirements of the IWE and IWL programs previously discussed, Technical Specification Surveillance Requirement 3.6.1.1 requires the performance of required visual examinations and leakage rate testing in accordance with the site's Containment Leakage Rate Testing Program.

American Society of Mechanical Engineers Section XI, 1992 Subsections IWE and IWL requires a visual examination of accessible interior and exterior surfaces of Containment be performed, for those sites where the interval for Type A test has been extended to ten years, prior to initiating a Type A test and during two other refueling outages before the next Type A test.

At Calvert Cliffs the above requirements are met through the performance of a surveillance test procedure. The purpose of this surveillance test is to perform a visual inspection of the normally accessible internal and exterior surfaces of the primary containment to identify evidence of structural deterioration, which might affect either the structural integrity or leak tightness of the Containment. Any condition identified as impacting either the structural integrity or leak tightness of Containment is documented and corrected.

This surveillance test procedures is currently performed each refueling outage as a result of a commitment made as part of the Calvert Cliffs license renewal application. The performance of this surveillance test every refueling outage exceeds the above mentioned requirements.

3.11 License Renewal Aging Management

The containment structures are in scope for license renewal based on 10 CFR 54.4(a).

Reference 19 lists the plausible age-related degradation mechanisms of the containment components. These age-related degradation mechanisms are managed through the conduct of various surveillance tests, in-service inspections, preventive maintenance activities, and maintenance procedures. These documents will continue to be modified as necessary to ensure they continue to provide reasonable assurance that the aging effects will be adequately managed throughout the operating life of the units.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements

4.1.1 10 CFR Part 50, Appendix J, Option B

The testing requirements of 10 CFR Part 50, Appendix J, provide assurance that the leakage from the primary containment, including systems and components that penetrate the Containment, does not exceed the allowable leakage values specified in the Technical Specification. This limitation on containment leakage provides assurance that the primary containment will continue to perform its design function following any plant design basis accidents. This appendix provides requirements for Types A, B and C testing.

Title 10 CFR Part 50, Appendix J was revised, effective October 26, 1995, to allow licensees to perform containment leakage testing in accordance with the requirements of Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." Technical Specification 5.5.16 implements Option B of 10 CFR Part 50, Appendix J as modified by NRC-approved exemptions. The performance-based ILRT requirements of Option B of 10 CFR Part 50, Appendix J, provide an alternative to the three tests per ten-year frequency specified by the prescriptive requirements of Option A of 10 CFR Part 50, Appendix J.

4.1.2 Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program"

Regulatory Guide 1.163 specifies an acceptable method for complying with the inspection and testing requirements of 10 CFR Part 50, Appendix J, Option B. Regulatory Position C.1 of Regulatory Guide 1.163 states that licensees should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 (Reference 5). Deviations to Regulatory Guide 1.163 are permitted by 10 CFR Part 50, Appendix J, Option B, as discussed in Section V.B.

4.1.3 Nuclear Energy Institute 94-01, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J"

This guideline provides direction for implementing the Option B testing and scheduling those tests to ensure compliance to the regulations. As documented in Regulatory Guide 1.163, the NRC has endorsed NEI 94-01 (Reference 5) as providing acceptable methods for complying with the requirements of Option B of 10 CFR Part 50, Appendix J. Nuclear Energy Institute 94-01 specifies an ILRT frequency of one test per ten years if certain performance criteria are met.

4.1.4 Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis"

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10-6/yr and increases in large early release frequency (LERF) below 10-7/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF.

4.2 Precedent

The NRC has approved one time extensions of the ILRT interval to 15 years based on risk and non-risk based considerations for other licensees including Waterford Steam Electric Station, Unit 3 (Reference 8), Calvert Cliffs Nuclear Power Plant, Unit 1 (Reference 9), Crystal River Nuclear Plant, Unit 3 (Reference 10), Indian Point 3 Nuclear Power Plant (Reference 11), D.C. Cook Nuclear Plant, Units 1 and 2 (Reference 12) and Palo Verde Units 1, 2, and 3 (Reference 13).

4.3 No Significant Hazards Consideration

The proposed amendment to the Calvert Cliffs Unit 2 Administrative Technical Specification 5.5.16 would add a one-time exception to the commitment to follow the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." The exception is based on information in Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J." The effect of this request will be an extension of the interval since the last Integrated Leakage Rate Test (ILRT) from 10 years to 15 years.

The proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to not involve a significant hazards consideration in that operation of the facility in accordance with the proposed amendment:

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

   No.

   This proposed one-time extension of the Type A test interval from 10 years to 15 years does not increase the probability of an accident since there are no design or operating changes involved and the test is not an accident initiator. The proposed extension of the test interval does not involve a significant increase in the consequences of an accident since research documented in NUREG-1493 has found that, generically, fewer than 3% of the potential containment leak paths are not identified by Types B and C testing. Calvert Cliffs, through testing and containment inspections, also provides a high degree of assurance that the Containment will not degrade in a manner detectable only by a Type A test. Inspections required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code are performed to identify containment degradation that could affect leak tightness.

   Therefore, this proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

   No.

   This proposed one-time extension of the Type A test interval from 10 years to 15 years does not involve any design or operational changes that could lead to a new or different kind of accident from any accident previously evaluated. The test itself is not changing and will be performed after a longer interval. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation.

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

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

   No.

   The proposed one-time extension of the Type A test interval from 10 years to 15 years does not involve a significant reduction in the margin of safety of the containment's ability to maintain its integrity during a design basis accident. The generic study of the increase in the Type A test interval, NUREG-1493, concluded there is an imperceptible increase in the plant risk associated with extending the test interval out to 20 years. Further, the extended test interval would have a minimal effect on this risk since Types B and C testing detect 97% of potential leakage paths. For the requested change in the Calvert Cliffs Integrated Leakage Rate Test interval, it was determined that the risk contribution of leakage will increase 0.07% (based on change in offsite dose). This change is considered very small and does not represent a significant reduction in the margin of safety.

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

Based on the above, Calvert Cliffs concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATIONS

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

6.0 REFERENCES

1. Regulatory Guide 1.174, dated July 1998, An Approach for, Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis
2. Regulatory Guide 1.200, Revision 2, dated March 2009, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities
3. NRC Information Notice 92-20, dated March 3, 1992, Inadequate Local Leak Rate Testing
4. Regulatory Guide 1.163, dated September 1995, Performance-Based Containment Leak Test Program
5. Nuclear Energy Institute document NEI 94-01, Revision 0, dated July 26, 1995. Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J
6. NUREG-1493, dated September 1995, Performance-Based Containment Leak-Test Program

7. Electric Power Research Institute report TR-104285, dated August 1994, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals

8. Letter from N. Kalyanam (NRC) to J. E. Venable (Entergy Operations Incorporated), dated February 14, 2002, Waterford Steam Electric Station, Unit 3 - Issuance of Amendment Re: Integrated Leakage Rate Testing Interval Extension (TAC No. MB2461) (ML020460272)

9. Letter from D. Skay (NRC) to C. H. Cruse (CCNPP), dated May 1, 2002, Calvert Cliffs Nuclear Power Plant, Unit No. 1 – Amendment Re: One-Time Extension of Appendix J, Type A, Integrated Leak Rate Test Interval And Exception From Performing A Post-Modification Type A Test (TAC No. MB3929) (ML021080753)

10. Letter from J. M. Goshen (NRC) to D. E. Young (Florida Power Corporation), dated August 30, 2001, Crystal River Unit 3 - Issuance of Amendment regarding Containment Leakage Rate Testing Program (TAC No. MB 1349) (ML012190219)

11. Letter from G. F. Wunder (NRC) to M. Kansler (Entergy Nuclear Operations Incorporated), dated April 17, 2001, Indian Point Nuclear Generating Unit 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB-0178) (ML011021315)

12. Letter from J. F. Stang (NRC) to A. C. Bakken III (Indiana Michigan Power Company), dated February 25, 2003, Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments (TAC Nos. MB4837 and MB4838) (ML030160330)

13. Letter from R. Hall (NRC) to R. Edington (Arizona Public Service), dated October 20, 2009, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Issuance of Amendments Re: Revision to Technical Specification 5.5.16, Containment Leakage Rate Testing Program (TAC Nos. MD9807, MD9808, and MD9809) (ML092810317)

14. Letter from C. H. Cruse (CCNPP) to Document Control Desk (NRC), dated November 19, 2001, License Amendment Request: Revision to the Containment Leakage Rate Testing Program Technical Specification to Support Steam Generator Replacement

15. Letter from D. Skay (NRC) to P. E. Katz (CCNPP), dated June 27, 2002, Calvert Cliffs Nuclear Power Plant, Unit 2 – Amendment Re: Exception to Post-Modification Integrated Leakage Rate Testing (TAC No. MB3444)

16. NEI 94-01, Revision 2, NRC approved June 25, 2008, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J

17. EPRI TR-1009325 Revision 2, Final Report, dated August 2007, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, which was approved by the NRC on June 25, 2008

18. Regulatory Guide 1.147, dated October 2007, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1

19. Calvert Cliffs Nuclear Power Plant, Updated Final Safety Analysis Report, Chapter 16, Revision 41, September 15, 2009

ENCLOSURE (1)

Unit 2 Type B and Type C LLRT Results

Enclosure 1

Calvert Cliffs Unit 2
As-Found / As-Left LLRT Results
RFO 2001 - RFO 2009

UNIT TWO															
Pen #	**Description**	**Valve (Test #)**	**Admin Limit**	**Max Limit**	**2001 AF**	**2001 AL**	**2003 AF**	**2003 AL**	**2005 AF**	**2005 AL**	**2007 AF**	**2007 AL**	**2009 AF**	**2009 AL**
1A	RCS and Pressurizing Sampling	CV-5464 (1)	111	2000	2	38.7	38	38	2		2	2	2	2
		CV-5465 (2)	111	2000	3	20	20	20	2		2	2	2	2
		CV-5466 (3)	111	2000	7	35.1	35.1	35.1	5.5		2	80	2	2
		CV-5467 (4)	111	2000	6	20	20	20	6.5		6.5	6.5	6.5	30
1B	RC drain tank vent header to	CV-2181 (1)	296	10000	41.5	41.5	93	87.3	87.3		34	34	34	34
		CV-2180 (2)	296	10000	120.5	120.5	91	193.4	193.4		85	85	85	85
1C	RCP seal bleed off	CV-505 (1)	111	2000	11.2	11.2	11.8	37.7	37.7		4.4	4.4	4.4	4.4
		CV-506 (2)	111	2000	20	20	2.5	2	2		2	2	2	2
1D	PASS return to RC drain tank	SV-6529	37	2000	20	20	2	2	2		2	2	2	2
2A	CVCS letdown	CV-516 (1)	296	10000	20	20	20	47	47		34.5	5	60	60
		CVC-103/105 (2)	296	10000	276	175	142	138	219	153.2	57	170	38	38
		CV-515 (3)	296	10000	171	171	20	37	37		24.5	2	80	80
2B	RC Charging	CV-184 (1)	296	10000	2	2	901	720	N/A	N/A	N/A	N/A	N/A	N/A
		CVC-435 (2)	296	10000	425	425	266	1488	N/A	N/A	N/A	N/A	N/A	N/A
		CV-517 (3)	296	10000	20	31.6	31.6	31.6	N/A	N/A	N/A	N/A	N/A	N/A
		CV-519 (4)	296	10000	20	20	20	20	N/A	N/A	N/A	N/A	N/A	N/A
		CV-518 (5)	296	10000	20	58.8	58.8	58.8	N/A	N/A	N/A	N/A	N/A	N/A
7A	ILRT test connection	Flange/ILRT-1	150	300	10.85	49.7	20	20	20		3.8	3.8	3.8	3.8
7B	ILRT test connection	Flange/ILRT-2	150	300	2	18.4	20	20	20		2	2	2	2
8	CNTMT Sump	MOV-5462	591	10000	98	20	176	22	N/A	N/A	N/A	N/A	N/A	N/A
		MOV-5463	591	10000	5310	20	174	174	N/A	N/A	N/A	N/A	N/A	N/A
9	Containment spray	SI-340 (1)	1182	20000	97	441	394	486	116	45	1072	1470	1470	820
		SI-326 (2)	1182	20000	62	62	20	127	127	50	145	145	145	145
10	Containment spray	SI-330 (1)	1182	20000	2750	250	438	311	950	10.5	730	880	650	650
		SI-316 (2)	1182	20000	112.3	39.4	20	570	570	780	780	780	510	510
13	Pen	Flange	250	500	2	20	2	20	20	2	2	2	2	2
14	Purge air inlet and outlet	Flange	250	500	2	20	2	20	20	2	2	2	2	2

Enclosure 1

Calvert Cliffs Unit 2
As-Found / As-Left LLRT Results
RFO 2001 - RFO 2009

UNIT TWO															
Pen #	Description	Valve (Test #)	Admin Limit	Max Limit	2001 AF	2001 AL	2003 AF	2003 AL	2005 AF	2005 AL	2007 AF	2007 AL	2009 AF	2009 AL
15	Containment atmosphere and	CV-5292 (1)	148	2000	26	26	26	26	15	15	15	15	15	40
		CV-5291 (2)	148	2000	20	20	20	25	25	25	5	5	5	5
16	CCW to RCPs	CV-3832	1478	20000	374	374	2	2	2	2	2	2	50	50
18	CCW from RCPs	CV-3833	1478	20000	8.6	8.6	35	35	10	10	10	10	370	370
19A	Instrument air to containment	IA-175 (1)	296	10000	20	20	20	20	10.5	10.5	2	2	2	2
		MOV-2080 (2)	296	10000	73.9	30.3	30	30	249	249	173	2	600	600
19B	Plant air	PA-137 (1)	296	10000	21	21	21	21	35	35	35	35	30	30
		PA-1044 (2)	296	10000	4.6	4.6	5	5	15	15	15	15	11	11
20A	Nitrogen Supply	CV612-642 and N2-347 (1)	148	2000	395	395	265	265	510	510	290	1129	600	450
		CV612-642 (2)	148	2000	395	395	266	266	510	510	280	1119	580	440
20B	Nitrogen Supply	N2-395 (1)	148	2000	40.1	40.1	2	2	30		14	14	2	2
		N2-348 (2)	148	2000	112	112	38.2	38.2	72		72	72	65	65
20C	Nitrogen Supply	N2-398 (1)	148	2000	54.9	54.9	2	2	42		42	42	2	2
		N2-349 (2)	148	2000	52	52	39.8	39.8	59		59	59	55	55
21SG (2HXRC21)	North Manway (1)	North Manway (1)	2500	5000	3.3	201	201	145	145	62	62	12	12	12
21 SG (2HXRC21)	South Manway (2)	South Manway (2)	2500	5000	2	161.8	161.8	1701	1701	80	80	19.8	19.8	57
22 SG (2HXRC22)	North Manway (1)	North Manway (1)	2500	5000	2	885	885	97	97	75	75	8.4	8.4	2
22 SG (2HXRC22)	South Manway (2)	South Manway (2)	2500	5000	2	121.9	121.9	2	2	76.2	76.2	2	2	2
23	Rx coolant drain tank drains	CV-4260	296	10000	38	81.2	40.5	40.5	40.5		710	10	7	7
24	Oxygen Sample Line	SV-6531	37	2000	6.5	6.5	2	2	2		65	65	2	2
37	Plant Service Water	PSW-1020 (1)	443	10000	91.5	91.5	91.5	91.5	2		2	2	350	350
		PSW-1009 (2)	443	10000	2	2	2	2	14		14	14	4	4
38	Demineralized Water	CV-5460	296	10000	36	36	285	17	17		20	190	77	77

Enclosure 1

Calvert Cliffs Unit 2
As-Found / As-Left LLRT Results
RFO 2001 - RFO 2009

UNIT TWO														
Pen #	Description	Valve (Test #)	Admin Limit	Max Limit	2001 AF	2001 AL	2003 AF	2003 AL	2005 AF	2005 AL	2007 AF	2007 AL	2009 AF	2009 AL
39	SIT Leakage Test	SI-455 (1)	296	10000	20	20	20	20	4		4	4	10	10
		SI-463 (2)	296	10000	6.5	6.5	6.5	6.5	62		62	62	90	90
41	Shutdown Cooling	MOV-651 and 652	1770	40000	4768	715	477	476.8	953.6		1430	476.8	2193	2193
42	Fuel Transfer Tube	Blind Flange	250	500	18	20	20	2	2	3.5	3.5	8	8	7
44	Fire Protection	FP-145A (1)	887	20000	55.4	55.4	55.4	55.4	17.5	112	310	310	300	300
		FP-145B (2)	887	20000	4.4	4.4	4.4	4.4	2	133	2	2	7480	744
47A	Hydrogen Sample	SV-6507A (1)	37	2000	8.53	8.53	2	2	2		3	3	3	3
		SV-6540A (2)	37	2000	20	20	2	2	2		4	4	4	3
47B	Hydrogen Sample	SV-6507E (1)	37	2000	45.2	45.2	2	2	20		20	20	2	2
		SV-6540E (2)	37	2000	3	3	3.6	3.6	3.6		6	6	6	6
47C	Hydrogen Sample	SV-6507F (1)	37	2000	54.5	54.5	2	2	20		20	20	2	2
		SV-6540F (2)	37	2000	2	2	2	2	2		2.5	2.5	2.5	2.5
47D	Hydrogen Sample	SV-6507G (1)	37	2000	29.5	29.5	2	2	2		2	2	2	2
		SV-6540G (2)	37	2000	2	2	2	2	2		2	2	2	2
48A	Hydrogen Purge	MOV-6901 (1)	591	10000	73.3	19.2	17	17	3.2		28	28	113	113
		MOV-6900 (2)	591	10000	22.3	20	20	20	2		43	43	60	60
48B	Hydrogen Purge Supply	MOV-6903 (1)	591	10000	248	248	117	117	104	169	2600	2600	2300	2300
		HP-104 (2)	591	10000	65	65	20	20	16		14	14	14	14
49A	Hydrogen Sample	SV-6507B (1)	37	2000	20	20	2	2	2		2	2	2	2
		SV-6540B (2)	37	2000	2	2	2	2	2		2	2	2	2
49B	Hydrogen Sample	SV-6507C (1)	37	2000	20	20	2	2	2		5	5	5	5
		SV-6540C (2)	37	2000	3	3	2	2	2		2	2	2	2
49C	Hydrogen Sample	SV-6507D (1)	37	2000	20	20	2	2	2		6	6	6	6
		SV-6540D (2)	37	2000	20	20	2	2	2		2	2	2	2
50	ILRT Pressurization	2 Blind Flanges	250	500	3.79	6.72	12.5	8	8	2	2	4.9	4.9	2

Enclosure 1

Calvert Cliffs Unit 2
As-Found / As-Left LLRT Results
RFO 2001 - RFO 2009

UNIT TWO

Pen #	Description	Valve (Test #)	Admin Limit	Max Limit	2001 AF	2001 AL	2003 AF	2003 AL	2005 AF	2005 AL	2007 AF	2007 AL	2009 AF	2009 AL
53	U-2 EAST Electrical Penetration	ZEA4	50	200	19	5	5	5	5		2	2	2	2
		ZEA5	50	200	5	2	2	2	2		2	2	2	2
		ZEB1	50	200	2	2	2	2	2		2	2	2	2
		ZEB2	50	200	2.1	2	2	2	2		2	2	2	2
		ZEB4	50	200	6.1	2	2	2	2		2	2	2	2
		ZEC1	50	200	8.7	15	15	15	15		2	2	2	2
		ZED8	50	200	5.1	2	2	2	2		2	2	2	2
		ZEE1	50	200	12.9	11	11	11	11		2	2	2	2
		ZEE3	50	200	5.7	2	2	2	2		11.6	11.6	11.6	11.6
		ZEE4	50	200	13	4	4	4	4		5.7	5.7	5.7	5.7
55	U-2 EAST Electrical Penetration	ZEA7	50	200	10.2	5	5	5	5		11	11	11	11
		ZEB3	50	200	9.7	2	2	2	2		2	2	2	2
		ZEB5	50	200	7.9	2	2	2	2		2.8	2.8	2.8	2.8
		ZEB6	50	200	5.4	2	2	2	2		15.4	15.4	15.4	15.4
		ZEC2	50	200	6.5	2	2	2	2		2	2	2	2
		ZEC4	50	200	2	2	2	2	2		5.2	5.2	5.2	5.2
		ZEC6	50	200	23	3	3	3	3		8.9	8.9	8.9	8.9
		ZEC7	50	200	2	2	2	2	2		2	2	2	2
		ZEC9	50	200	10.6	10	10	10	10		2.7	2.7	2.7	2.7
		ZED1	50	200	2	2	2	2	2		2	2	2	2
		ZED2	50	200	2	2	2	2	2		2	2	2	2
		ZED3	50	200	2.6	4	4	4	4		8.5	8.5	8.5	8.5
		ZED4	50	200	2.6	8	8	8	8		2.9	2.9	2.9	2.9
		ZED5	50	200	3.9	3	3	3	3		2.3	2.3	2.3	2.3
		ZED6	50	200	11.8	2	2	2	2		2	2	2	2
		ZED7	50	200	2	2	2	2	2		2	2	2	2
		ZEE2	50	200	4.9	4	4	4	4		14.7	14.7	14.7	14.7
		ZEE5	50	200	13.67	14	14	14	14		5.9	5.9	5.9	5.9
		ZEE6	50	200	7.01	7	7	7	7		2.5	2.5	2.5	2.5
		ZEE7	50	200	11.45	11	11	11	11		5.3	5.3	5.3	5.3
		ZEE8	50	200	12.99	13	13	13	13		4.5	4.5	4.5	4.5
		ZEE9	50	200	8.52	9	9	9	9		4.4	4.4	4.4	4.4

Enclosure 1

Calvert Cliffs Unit 2
As-Found / As-Left LLRT Results
RFO 2001 - RFO 2009

UNIT TWO

Pen #	Description	Valve (Test #)	Admin Limit	Max Limit	2001 AF	2001 AL	2003 AF	2003 AL	2005 AF	2005 AL	2007 AF	2007 AL	2009 AF	2009 AL
56	U-2 EAST Electrical	ZEA2	50	200	2	2	2	2	2		2	2	2	2
		ZEA9	50	200	6.1	7	7	7	7		3.7	3.7	3.7	3.7
54	U-2 WEST Electrical Penetration	ZWA3	50	200	29	29	29	29	29		9.1	9.1	9.1	9.1
		ZWB1	50	200	6	6	6	6	6		10.3	10.3	10.3	10.3
		ZWB2	50	200	2	2	2	2	2		2.3	2.3	2.3	2.3
		ZWB8	50	200	2	2	2	2	2		2	2	2	2
		ZWC1	50	200	2	2	2	2	2		6.9	6.9	6.9	6.9
		ZWC3	50	200	2	2	2	2	2		2	2	2	2
		ZWC4	50	200	2	2	2	2	2		4.4	4.4	4.4	4.4
		ZWD1	50	200	3	3	3	3	3		2.1	2.1	2.1	2.1
		ZWD2	50	200	14	14	14	14	14		34	34	34	34
		ZWE1	50	200	4	4	4	4	4		4.6	4.6	4.6	4.6
		ZWE3	50	200	3	3	3	3	3		10.3	10.3	10.3	10.3
		ZWE6	50	200	22	8	8	8	8		11.4	11.4	11.4	11.4
55	U-2 WEST Electrical Penetration	ZWA6	50	200	48	48	48	48	48		10	10	10	10
		ZWB7	50	200	2	2	2	2	2		10.5	10.5	10.5	10.5
		ZWC6	50	200	4	4	4	4	4		11.6	11.6	11.6	11.6
		ZWC9	50	200	5	5	5	5	5		14.5	14.5	14.5	14.5
		ZWD3	50	200	7	7	7	7	7		10.2	10.2	10.2	10.2
		ZWD4	50	200	3	3	3	3	3		17	17	17	17
		ZWD5	50	200	7	7	7	7	7		7	7	7	7
		ZWD6	50	200	2	2	2	2	2		2	2	2	2
		ZWD7	50	200	15	15	15	15	15		5.1	5.1	5.1	5.1
		ZWD8	50	200	2	2	2	2	2		2.9	2.9	2.9	2.9
		ZWD9	50	200	7	7	7	7	7		5.9	5.9	5.9	5.9
		ZWE2	50	200	2	2	2	2	2		5.8	5.8	5.8	5.8
		ZWE4	50	200	4	4	4	4	4		2.8	2.8	2.8	2.8
		ZWE5	50	200	2	2	2	2	2		2	2	2	2
		ZWE7	50	200	9	9	9	9	9		3.8	3.8	3.8	3.8
		ZWE8	50	200	10	10	10	10	10		13.9	13.9	13.9	13.9
		ZWE9	50	200	2	2	2	2	2		10.8	10.8	10.8	10.8

Enclosure 1

Calvert Cliffs Unit 2
As-Found / As-Left LLRT Results
RFO 2001 - RFO 2009

UNIT TWO														
Pen #	**Description**	**Valve (Test #)**	**Admin Limit**	**Max Limit**	**2001 AF**	**2001 AL**	**2003 AF**	**2003 AL**	**2005 AF**	**2005 AL**	**2007 AF**	**2007 AL**	**2009 AF**	**2009 AL**
56	U-2 WEST Electrical	ZWA1	50	200	2	2	2	2	2		4.4	4.4	4.4	4.4
		ZWA8	50	200	2	2	2	2	2		16.5	16.5	16.5	16.5
59	SFP Cooling to Refueling Pool	SFP-178/179	1182	20000	45	14.8	30.8	30.8	30.8		82	20	20	2
60	Auxiliary Steam	ES-142 and 144	148	2000	32.3	32.3	32.3	32.3	3.6		3.6	3.6	14	14
61	SFP Cooling from Refueling Pool	SFP-180, 182, 184, 186	1182	20000	53.2	299	299	299	3250	1405	960	960	5000	86
67	Equipment Hatch	Equip Hatch	200	400	6	20	20	2	2	22	22	2	2	2
68	Personal Air Lock	PAL	10000	17300	5547.9	5547.9	3699	3699	4808		3698.6	3698.6	3698.6	
69	Emergency Airlock	EAL	10000	17300	2773.9	2773.9	3051	3051	1479.4		1479.4	924.7	924.7	2127
84	ILRT Vent	FLANGE	250	500	N/A	10.87	3	3	3	3	3	3	3	3

ENCLOSURE (2)

Unit 2 Type B and Type C LLRT Schedule

Enclosure 2

Calvert Cliffs Unit 2
Type B and Type C LLRT Schedule
RFO-2011 - RFO-2017

Pen #	Description	Valve (Test #)	Admin Limit	Max Limit	2011 RFO Y/N	2013 RFO Y/N	2015 RFO Y/N	2017 RFO Y/N
1A	RCS and Pressurizing Sampling	CV-5464 (1)	111	2000	N	Y	Y	N
		CV-5465 (2)	111	2000	N	Y	Y	N
		CV-5466 (3)	111	2000	N	Y	Y	N
		CV-5467 (4)	111	2000	Y	Y	Y	N
1B	RC drain tank vent header to waste gas	CV-2181 (1)	296	10000	Y	N	Y	Y
		CV-2180 (2)	296	10000	Y	N	Y	Y
1C	RCP seal bleed off	CV-505 (1)	111	2000	Y	N	Y	Y
		CV-506 (2)	111	2000	Y	N	Y	Y
1D	PASS return to RC drain tank	SV-6529	37	2000	Y	N	Y	Y
2A	CVCS letdown	CV-516 (1)	296	10000	N	Y	Y	N
		CVC-103/105 (2)	296	10000	N	Y	Y	N
		CV-515 (3)	296	10000	N	Y	Y	N
2B	RC Charging	CV-184 (1)	296	10000	NA	NA	NA	NA
		CVC-435 (2)	296	10000	NA	NA	NA	NA
		CV-517 (3)	296	10000	NA	NA	NA	NA
		CV-519 (4)	296	10000	NA	NA	NA	NA
		CV-518 (5)	296	10000	NA	NA	NA	NA
7A	ILRT test connection	Flange/ILRT-1	150	300	Y	N	Y	N
7B	ILRT test connection	Flange/ILRT-2	150	300	Y	N	Y	N
8	CNTMT Sump	MOV-5462	591	10000	NA	NA	NA	NA
		MOV-5463	591	10000	NA	NA	NA	NA
9	Containment spray	SI-340 (1)	1182	20000	Y	Y	Y	N
		SI-326 (2)	1182	20000	Y	N	Y	N
10	Containment spray	SI-330 (1)	1182	20000	N	Y	Y	N
		SI-316 (2)	1182	20000	N	Y	Y	N
13	Pen	Flange	250	500	Y	Y	Y	Y
14	Purge air inlet and outlet	Flange	250	500	Y	Y	Y	Y
15	Containment atmosphere and purge	CV-5292 (1)	148	2000	Y	Y	Y	N
		CV-5291 (2)	148	2000	Y	N	Y	N
16	CCW to RCPs	CV-3832	1478	20000	N	Y	Y	N
18	CCW from RCPs	CV-3833	1478	20000	N	Y	Y	N
19A	Instrument air to containment	IA-175 (1)	296	10000	N	Y	Y	N
		MOV-2080 (2)	296	10000	Y	Y	Y	N
19B	Plant air	PA-137 (1)	296	10000	N	Y	Y	N
		PA-1044 (2)	296	10000	N	Y	Y	N
20A	Nitrogen Supply	CV612-642 and N2-347 (1)	148	2000	Y	Y	Y	N
		CV612-642 (2)	148	2000	Y	Y	Y	N

Enclosure 2

Calvert Cliffs Unit 2

Type B and Type C LLRT Schedule

RFO-2011 - RFO-2017

Pen #	Description	Valve (Test #)	Admin Limit	Max Limit	2011 RFO Y/N	2013 RFO Y/N	2015 RFO Y/N	2017 RFO Y/N
20B	Nitrogen Supply	N2-395 (1)	148	2000	N	Y	Y	N
		N2-348 (2)	148	2000	N	Y	Y	N
20C	Nitrogen Supply	N2-398 (1)	148	2000	N	Y	Y	N
		N2-349 (2)	148	2000	N	Y	Y	N
21SG (2HXRC21)	North Manway (1)	North Manway (1)	2500	5000	Y	Y	Y	Y
21 SG (2HXRC21)	South Manway (2)	South Manway (2)	2500	5000	Y	Y	Y	Y
22 SG (2HXRC22)	North Manway (1)	North Manway (1)	2500	5000	Y	Y	Y	Y
22 SG (2HXRC22)	South Manway (2)	South Manway (2)	2500	5000	Y	Y	Y	Y
23	Rx coolant drain tank drains	CV-4260	296	10000	Y	Y	Y	Y
24	Oxygen Sample Line	SV-6531	37	2000	Y	Y	Y	Y
37	Plant Service Water	PSW-1020 (1)	443	10000	N	Y	Y	N
		PSW-1009 (2)	443	10000	N	Y	Y	N
38	Demineralized Water	CV-5460	296	10000	Y	Y	Y	N
39	SIT Leakage Test	SI-455 (1)	296	10000	N	Y	Y	N
		SI-463 (2)	296	10000	N	Y	Y	N
41	Shutdown Cooling	MOV-651 and 652	1770	40000	Y	Y	Y	N
42	Fuel Transfer Tube	Blind Flange	250	500	Y	Y	Y	Y
44	Fire Protection	FP-145A (1)	887	20000	Y	Y	Y	Y
		FP-145B (2)	887	20000	Y	Y	Y	Y
47A	Hydrogen Sample	SV-6507A (1)	37	2000	Y	N	Y	Y
		SV-6540A (2)	37	2000	Y	N	Y	Y
47B	Hydrogen Sample	SV-6507E (1)	37	2000	N	Y	Y	N
		SV-6540E (2)	37	2000	Y	N	Y	Y
47C	Hydrogen Sample	SV-6507F (1)	37	2000	N	Y	Y	N
		SV-6540F (2)	37	2000	Y	N	Y	Y
47D	Hydrogen Sample	SV-6507G (1)	37	2000	Y	N	Y	Y
		SV-6540G (2)	37	2000	Y	N	Y	Y
48A	Hydrogen Purge	MOV-6901 (1)	591	10000	Y	Y	Y	Y
		MOV-6900 (2)	591	10000	Y	Y	Y	Y
48B	Hydrogen Purge Supply	MOV-6903 (1)	591	10000	Y	Y	Y	N
		HP-104 (2)	591	10000	Y	N	Y	Y
49A	Hydrogen Sample	SV-6507B (1)	37	2000	Y	N	Y	Y
		SV-6540B (2)	37	2000	Y	N	Y	Y
49B	Hydrogen Sample	SV-6507C (1)	37	2000	Y	N	Y	Y
		SV-6540C (2)	37	2000	Y	N	Y	Y
49C	Hydrogen Sample	SV-6507D (1)	37	2000	Y	N	Y	Y
		SV-6540D (2)	37	2000	Y	N	Y	Y
50	ILRT Pressurization	2 Blind Flanges	250	500	Y	Y	Y	Y

Enclosure 2

Calvert Cliffs Unit 2
Type B and Type C LLRT Schedule
RFO-2011 - RFO-2017

Pen #	Description	Valve (Test #)	Admin Limit	Max Limit	2011 RFO Y/N	2013 RFO Y/N	2015 RFO Y/N	2017 RFO Y/N
53	U-2 EAST Electrical Penetration	ZEA4	50	200	N	Y	Y	Y
		ZEA5	50	200	N	Y	Y	Y
		ZEB1	50	200	N	Y	Y	Y
		ZEB2	50	200	N	Y	Y	Y
		ZEB4	50	200	N	Y	Y	Y
		ZEC1	50	200	N	Y	Y	Y
		ZED8	50	200	N	Y	Y	Y
		ZEE1	50	200	N	Y	Y	Y
		ZEE3	50	200	N	Y	Y	Y
		ZEE4	50	200	N	Y	Y	Y
55	U-2 EAST Electrical Penetration	ZEA7	50	200	N	Y	Y	N
		ZEB3	50	200	N	Y	Y	N
		ZEB5	50	200	N	Y	Y	N
		ZEB6	50	200	N	Y	Y	N
		ZEC2	50	200	N	Y	Y	N
		ZEC4	50	200	N	Y	Y	N
		ZEC6	50	200	N	Y	Y	N
		ZEC7	50	200	N	Y	Y	N
		ZEC9	50	200	N	Y	Y	N
		ZED1	50	200	N	Y	Y	N
		ZED2	50	200	N	Y	Y	N
		ZED3	50	200	N	Y	Y	N
		ZED4	50	200	N	Y	Y	N
		ZED5	50	200	N	Y	Y	N
		ZED6	50	200	N	Y	Y	N
		ZED7	50	200	N	Y	Y	N
		ZEE2	50	200	N	Y	Y	N
		ZEE5	50	200	N	Y	Y	N
		ZEE6	50	200	N	Y	Y	N
		ZEE7	50	200	N	Y	Y	N
		ZEE8	50	200	N	Y	Y	N
		ZEE9	50	200	N	Y	Y	N
56	U-2 EAST Electrical Penetration	ZEA2	50	200	N	Y	Y	N
		ZEA9	50	200	N	Y	Y	N

Enclosure 2

Calvert Cliffs Unit 2
Type B and Type C LLRT Schedule
RFO-2011 - RFO-2017

Pen #	Description	Valve (Test #)	Admin Limit	Max Limit	2011 RFO Y/N	2013 RFO Y/N	2015 RFO Y/N	2017 RFO Y/N
54	U-2 WEST Electrical Penetration	ZWA3	50	200	N	Y	Y	N
		ZWB1	50	200	N	N	Y	N
		ZWB2	50	200	N	N	Y	N
		ZWB8	50	200	N	N	Y	N
		ZWC1	50	200	N	N	Y	N
		ZWC3	50	200	N	N	Y	N
		ZWC4	50	200	N	N	Y	N
		ZWD1	50	200	N	N	Y	N
		ZWD2	50	200	N	N	Y	N
		ZWE1	50	200	N	N	Y	N
		ZWE3	50	200	N	N	Y	N
		ZWE6	50	200	N	N	Y	N
55	U-2 WEST Electrical Penetration	ZWA6	50	200	N	N	Y	N
		ZWB7	50	200	N	N	Y	N
		ZWC6	50	200	N	N	Y	N
		ZWC9	50	200	N	N	Y	N
		ZWD3	50	200	N	N	Y	N
		ZWD4	50	200	N	N	Y	N
		ZWD5	50	200	N	N	Y	N
		ZWD6	50	200	N	N	Y	N
		ZWD7	50	200	N	N	Y	N
		ZWD8	50	200	N	N	Y	N
		ZWD9	50	200	N	N	Y	N
		ZWE2	50	200	N	N	Y	N
		ZWE4	50	200	N	N	Y	N
		ZWE5	50	200	N	N	Y	N
		ZWE7	50	200	N	N	Y	N
		ZWE8	50	200	N	N	Y	N
		ZWE9	50	200	N	N	Y	N
56	U-2 WEST Electrical Penetration	ZWA1	50	200	N	N	Y	N
		ZWA8	50	200	N	N	Y	N
59	SFP Cooling to Refueling Pool	SFP-178/179	1182	20000	Y	Y	Y	Y
60	Auxiliary Steam	ES-142 and 144	148	2000	N	Y	Y	N
61	SFP Cooling from Refueling Pool	SFP-180, 182, 184, 186	1182	20000	Y	Y	Y	N
67	Equipment Hatch	Equip Hatch	200	400	Y	Y	Y	Y
68	Personal Air Lock	PAL	10000	17300	Y	N	Y	N
69	Emergency Airlock	EAL	10000	17300	Y	Y	Y	N
84	ILRT Vent	FLANGE	250	500	N	N	Y	Y

ATTACHMENT (2)

MARKED UP TECHNICAL SPECIFICATION PAGE

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program

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

a. Nuclear Energy Institute (NEI) 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007.

INSERT

The first UNIT 2 Type A test performed after the May 2, 2001 Type A test shall be performed no later than May 1, 2016.

b. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

c. Unit 2 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, is 49.4 psig. The containment design pressure is 50 psig.

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

Leakage rate acceptance criteria are:

a. 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 criterion are $\leq 0.60\ L_a$ for Types B and C tests and $\leq 0.75\ L_a$ for Type A tests.

ATTACHMENT (3)

RISK ASSESSMENT OF THE PROPOSED AMENDMENT

RSC 10-21

CENG

a joint venture of

Constellation Energy

eDF

Calvert Cliffs Nuclear Power Plant Unit 2 Evaluation of Risk Significance of ILRT Extension

Revision 0

August 2010

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1				
2				
3				
4				
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1.0 PURPOSE

The purpose of this report is to provide an estimation of the change in risk associated with extending the Type A integrated leak rate test interval beyond the current 10 years specified by 10 CFR 50, Appendix J, Option B for Calvert Cliffs Nuclear Power Plant Unit 2. This activity supports a request for an exemption from the performance of the ILRT during the planned winter 2011 outage. The assessment is consistent with and processes described in the methodology identified in Reference 1.

1.1 SUMMARY OF THE ANALYSIS

10 CFR 50, Appendix J[2] allows individual plants to extend Type A surveillance testing requirements and to provide for performance-based leak testing. This report documents a risk-based evaluation of the proposed change of the integrated leak rate test (ILRT) interval for the Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 2. The proposed change would impact testing associated with the current surveillance tests for Type A leakage, procedure STP M-662-2[3]. No change to Type B or Type C testing is proposed at this time.

This analysis utilizes the methodology presented in Reference 1, the guidelines set forth in NEI 94-01[4], the approach presented in NUREG-1493[5] and considers the submittals generated by other utilities to define the scope of the analysis.

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

- 3 years – Interval based on the original requirements of 3 tests per 10 years.
- 10 years – This is the current test interval required for CCNPP Unit 2.
- 15 years – Proposed extended test interval.

To support the analysis, the analysis draws from the results of the current CCNPP Unit 2 probabilistic risk analysis (PRA) results[6]. The release category and person-rem information is based on the approach presented in Reference 1 and is based on plant-specific assessments for containment performance and offsite dose.

1.2 SUMMARY OF INTERNAL RESULTS/CONCLUSIONS

The specific results are summarized in Table 1 below. The Type A contribution to LERF is defined as the contribution from Class 3b.

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

	Risk Impact for 3-years (baseline)	Risk Impact for 10-years (current requirement)	Risk Impact for 15-years
Total integrated risk (person-rem/yr)	35.81	35.90	35.96
Type A testing risk (person-rem/yr)	4.15E-2	1.38E-1	2.08E-1
% total risk (Type A / total)	0.116%	0.385%	0.577%
Type A LERF (Class 3b) (per year)	6.47E-9	2.16E-8	3.23E-8
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			6.54E-2
% Increase from current (Δ Risk / Total Risk)			0.182%
Δ LERF from current (per year)			1.08E-8
Δ CCFP from current			4.35E-3
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			1.57E-1
% Increase from baseline (Δ Risk / Total Risk)			0.438%
Δ LERF from baseline (per year)			2.59E-8
Δ CCFP from baseline			1.04E-2

The results are discussed below:

- The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current once-per-ten-year interval to once-per-fifteen years is 6.54E-2 person-rem/year.
- The risk increase in LERF from extending the ILRT test frequency from the current once-per-10-year interval to once-per-15 years is 1.08E-8/yr.
- The change in conditional containment failure probability (CCFP) from the current once-per-10-year interval to once-per-15 years is 4.35E-3.
- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the risk impact on the total integrated plant risk by only 0.182%. Also, the change in Type A test frequency from the original three-per-ten-years to once-per-fifteen-years increases the risk only 0.438%. Therefore, the risk impact when compared to other severe accident risks is negligible.
- Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once per-fifteen-years is 1.08E-8/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1.0^{-7}/yr, increasing the ILRT interval from 10 to 15 years is therefore considered non-risk significant and the results support this determination. In addition, the change in LERF resulting from a change in the Type A ILRT test interval from a three-per-ten-years to a once per-fifteen-years is 2.59E-8/yr, is also below the guidance.
- R.G. 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 4.35E-3 for the proposed change and 1.04E-2 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. These changes are small and demonstrate that the defense-in-depth philosophy is maintained.

In reviewing these results the CCNPP Unit 2 analysis demonstrates that the change in plant risk is small as a result of this proposed extension of ILRT testing. The change in LERF defined in the analysis for both the baseline and the current cases is within the acceptance criterion.

In addition to the baseline assessment, three sensitivity exercises are included. These analyses are provided in Section 5 and are consistent with those outlined in Reference 1.

2.0 DESIGN INPUTS

The CCNPP Unit 2 PRA provides "best estimate" results that can be used as input when making risk informed decisions and is updated on a periodic basis. The inputs for this calculation come from the information documented in the CCNPP Unit 2 PRA Level 2 quantification notebook (Reference 6). The CCNPP Unit 2 release states are summarized in Table 2. CCNPP Unit 2 Level 2 results are lumped into 4 sequence states that represent the summation of individual accident categories. The number of sequences comprising each sequence state is also presented in Table 2.

Table 2
Release Category Frequencies

Release Category	Contributing CCNPP Unit 2 Accident Categories	Frequency (/yr)	EPRI Category
INTACT (S)	10	5.16E-06	Class 1
LERF	18	1.61E-06	Class 8
SERF	9	7.96E-08	Class 6
LATE	14	7.25E-06	Class 1[1]
Total	n/a	1.41E-05	n/a

1. Consistent with Reference 1 and based on the timing and mode of failure, contributions from late release category are classified as Class 1.

The LERF contribution for CCNPP Unit 2 contains bypass sequences, early containment failures due to containment phenomenon and due to large isolation failures. Consistent with the EPRI guidance these are parsed to Class 8, Class 7 and Class 2 respectively consistent with Reference 1.

Reference 6 provides the containment event tree endstate and associated frequency for each assessed sequence. Table 3 presents the sequence description, frequency and EPRI category for each sequence and the totals of each EPRI classification. Grouping each sequence endstate is based on the associated description.

Table 3
Decomposition of CCNPP Unit 2 LERF Frequencies and EPRI Classification

Endstate	Description of Outcome	Frequency (per year)	EPRI Category
LERF01	Containment failure following high-pressure (HP) vessel breach (VB)	4.30E-7	7
LERF02	Containment failure following HP VB	5.60E-9	7
LERF03	Containment failure following low pressure (LP) VB	1.98E-8	7
LERF04	Temperature induced (TI) SGTR	2.26E-10	8
LERF05	Containment failure following LP VB	3.66E-7	7
LERF06	Pressure induced (PI) SGTR	0	8
LERF07	Containment failure following LP VB	9.28E-9	7
LERF08	Loss of isolation	4.92E-8	2
LERF09	Containment bypass	6.19E-7	8
LERF10	Containment failure following LP VB	4.37E-8	7
LERF11	Containment failure following HP VB	8.68E-10	7
LERF12	Containment failure following LP VB	2.33E-9	7
LERF13	TI-SGTR	3.05E-10	8
LERF14	Containment failure following LP VB	4.84E-8	7
LERF15	PI—SGTR	0	8
LERF16	Containment failure following LP VB	0	7
LERF17	Loss of isolation	1.25E-8	2
LERF18	Containment bypass	6.33E-10	8
Contribution to EPRI Classification 2		4.90E-8	
Contribution to EPRI Classification 7		9.26E-7	
Contribution to EPRI Classification 8		6.20E-7	
Total LERF		1.61E-6	

The release category frequencies and EPRI classes are presented along with the associated person-rem doses for the EPRI classes are displayed in Table 4[7]. Class 1 consists of INTACT and LATE failures. Class 2 consists of the portion of LERF associated with large isolation failures.

One consideration in assigning frequency to the EPRI classes is that the level 2 model contains a bounding contribution associated with pre-event containment liner failure. To preclude influencing the current detailed assessment, the contribution associated with this failure (frequency contribution of 1.27E-8/yr) is adjusted. This involves removal of the bounding estimate from Class 2 category to the intact containment case (Class 1).

Class 6 is composed of the SERF contribution. Class 7 includes phenomenological failures. Class 8 retains the remaining portion of LERF which is related to bypass and steam generator tube rupture (SGTR) failures.

Table 4
CCNPP Unit 2 Dose for EPRI Accident Classes

Release Category	Frequency (/yr)	EPRI Class	CCNPP Unit 2 Dose (person-rem)
INTACT + LATE[1]	1.24E-5	Class 1	3.20E+4
LERF[2]	4.90E-8	Class 2	2.00E+7
SERF[3]	7.96E-8	Class 6	7.01E+6
LERF[4]	9.26E-7	Class 7	5.61E+7
LERF[5]	6.20E-7	Class 8	2.25E+7

1. The EPRI Class 1 category consists of INTACT and LATE failures. A LATE failure is classified as intact due to the long time until failure and is consistent with guidance in Reference 1.
2. The EPRI Class 2 category consists of CCNPP Unit 2 assigned LERF contribution associated with isolation failures as re-categorized in Table 3.
3. The EPRI Class 6 category consists of CCNPP Unit 2 assigned scrubbed isolation failures in SERF.
4. The EPRI Class 7 category consists of the CCNPP Unit 2 assigned LERF contribution associated with phenomenological failures as re-categorized in Table 3.
5. The EPRI Class 8 category consists of the CCNPP Unit 2 assigned LERF contribution associated with bypass or SGTR failures as re-categorized in Table 3.

3.0 ASSUMPTIONS

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

2. The maximum containment leakage for Class 3a (Reference 1) sequences is 10 L_a based on the EPRI guidance (Reference 1).

3. The maximum containment leakage for Class 3b sequences is 100 L_a based on the NEI guidance (Reference 1).

4. Class 3b is conservatively categorized as LERF based on the NEI guidance and previously approved methodology (Reference 1).

5. Containment leakage due to EPRI Classes 4 and 5 are considered negligible based on the NEI guidance and the previously approved methodology (Reference 1).

6. The containment releases are not impacted with time.

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

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

4.0 CALCULATIONS

This calculation applies the CCNPP Unit 2 PRA release category information in terms of frequency and person-rem estimates to estimate the changes in risk due to increasing the ILRT test interval. The changes in risk are assessed consistent with the guidance provided in the EPRI guidance document (Reference 1).

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

4.1 CALCULATIONAL STEPS

The analysis employs the steps provided in Reference 1 and uses associated risk metrics to evaluate the impact of a proposed change on plant risk. These measures are the change in release frequency, the change in risk as defined by the change in person-rem, the change in LERF and the change in the conditional containment failure probability.

Reference 1 also lists the change in core damage frequency as a measure to be considered. Since the testing addresses the ability of the containment to maintain its function, the proposed change has no measurable impact on core damage frequency. Therefore, this attribute remains constant and has no risk significance.

The overall analysis process is documented as outlined below:

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

- Compare analysis metrics to estimate the impact and significance of the increase related to those metrics.

The first step in the analysis is to define the baseline plant damage classes and person-rem dose measures. Plant damage state information is developed using the CCNPP Unit 2 PRA Level 2 PRA results. The containment end state information and the results of the containment analysis are used to define the representative sequences. The population person-rem dose values are provided in Reference 7 and were used in accordance to the guidance from Reference 1.

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

The PRA plant damage state definitions considered isolation failures due to Type B and Type C faults and examined containment challenges occurring after core damage and/or reactor vessel failure. These sequences are grouped into key plant damage classes. Using the plant damage state information, bypass, isolation failures and phenomena-related containment failures are identified. Once identified, the sequence was then classified by release category definitions specified in Reference 1. With this information developed, the PRA baseline inputs are completed.

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

Reference 1 provides the estimate for the probability of a leakage contribution that could only be identified by Type A testing based on industry experience. This probability is then used to adjust the intact containment category of the CCNPP Unit 2 PRA to develop a baseline model including Type A faults.

The release, in terms of person-rem, is developed based on information contained in Reference 1 and is estimated as a leakage increase relative to allowable dose (L_a) defined as part of the ILRT.

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

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

Using this process, the following were performed:

1. Map the CCNPP Unit 2 release categories into the 8 release classes defined by the EPRI Report (Reference 1).
2. Calculate the Type A leakage estimate to define the analysis baseline.
3. Calculate the Type A leakage estimate to address the current testing frequency.
4. Modify the Type A leakage estimates to address extension of the Type A test interval.
5. Calculate increase in risk due to extending Type A testing intervals.
6. Estimate the change in LERF due to the Type A testing.
7. Estimate the change in conditional containment failure probability due to the Type A testing.

4.2 SUPPORTING CALCULATIONS

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

Reference 1 defines eight (8) release classes as presented in Table 5.

Table 5
Containment Failure Classifications (from Reference 1)

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

Table 6 presents the CCNPP Unit 2 release category mapping for these eight accident classes. Person-rem per year is the product of the frequency (per year) and the person-rem.

Table 6
CCNPP Unit 2 PRA Release Category Grouping to EPRI Classes (Described in Reference 1)

Class	EPRI Description	Frequency	Person-Rem	Person-Rem/yr
1	Intact containment	1.24E-5[1]	3.20E+4	3.975E-1
2	Large containment isolation failures	4.90E-8	2.00E+7	9.80E-1
3a	Small isolation failures (liner breach)	Required		0.00E+0
3b	Large isolation failures (liner breach)	Required		0.00E+0
4	Small isolation failures - failure to seal (type B)	-		
5	Small isolation failures - failure to seal (type C)	-		
6	Containment isolation failures (dependent failure, personnel errors)	7.96E-8	7.01E+6	5.580E-1
7	Severe accident phenomena induced failure (early)	9.26E-7	5.61E+7	5.195E+1
8	Containment bypass	6.20E-7	2.25E+7	1.395E+1
	Total	1.41E-5		6.784E+1

1. The late contribution involves late failure. Consistent with guidance provided in Reference 1, this contribution is classified as Class 1.

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

As displayed in Table 6 the CCNPP Unit 2 PRA did not identify any release categories specifically associated with EPRI Classes 4 or 5 and Class the estimate for Class 3 was redistributed back into INTACT. Therefore each of these classes must be evaluated for applicability to this study.

Class 3:

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

In order to better assess the range of possible leakage rates, the Class 3 calculation is divided into two classes. Class 3a is defined as a small liner breach and Class 3b is defined as a large liner breach. This division is consistent with the EPRI guidance (Reference 1). The calculation of Class 3a and Class 3b probabilities is presented below.

Calculation of Class 3a Probability

The data presented in NUREG-1493 (Reference 5) is also used to calculate the probability that a liner leak will be small (Class 3a). The data found in NUREG-1493 states that 144 ILRTs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of $1.0L_a$. However, of the 23 events exceeding test requirements, only 4 were found by an ILRT, the others were found by Type B and C testing or were identified as errors in test alignments.

Data presented in Reference 1, taken since 1/1/1995, increases this database to a total of 5 Type A leakage events in total of 182 events. Using the data a mean estimate for the probability of leakage is determined for Class 3a as shown in Equation 1.

$$p_{Class3a} = \frac{5}{182} = 0.0275 \qquad \text{(eq. 1)}$$

This probability, however, is based on three tests over a 10-year period and not the one per ten-year frequency currently employed at CCNPP Unit 2 (Reference 3). The probability (0.0275) must be adjusted to reflect this difference and is adjusted in step 3 of this calculation.

Multiplying the CDF times the probability of a Class 3a leak develops the Class 3a frequency contribution in accordance with guidance provided in Reference 1. The total CDF includes contributions already binned to LERF. To include these contributions would result in a potentially conservative result. Therefore, the LERF contribution from CDF is removed (6.20E-7/yr). The CDF for CCNPP Unit 2 is 1.41E-5/yr as presented in Table 6 and is adjusted to remove the LERF contribution.

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

$FREQ_{class3a} = PROB_{class3a}$ x (CDF – Class 8)

$$= 0.0275 \times (1.41E\text{-}5/yr - 6.20E\text{-}7/yr) = 3.70E\text{-}7/yr \qquad \text{(eq. 2)}$$

Calculation of Class 3b Probability

To calculate the probability that a liner leak will be large (Class 3b) use was made of the data presented in the calculation of Class 3a. Of the events identified in NUREG-1493 (Reference 5), the largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Since 21 L_a does not constitute a large release, no large releases have occurred based on the 144 ILRTs reported in NUREG-1493. The additional data point was also not considered to constitute a large release.

To estimate the failure probability given that no failures have occurred, the guidance provided in Reference 1 suggests the use of a non-informative prior. This approach essentially updates a uniform distribution (no bias) with the available evidence (data) to provide a better estimation of an event.

A beta distribution is typically used for the uniform prior with the parameters α=0.5 and β=1. This is then combined with the existing data (no Class 3b events, 182 tests) using Equation 3.

$$p_{Class3b} = \frac{n+\alpha}{N+\beta} = \frac{0+0.5}{182+1} = \frac{0.5}{183} = 0.00273 \qquad \text{(eq. 3)}$$

where: N is the number of tests, n is the number of events (faults) of interest, α, β are the parameters of the non-informative prior distribution. From this solution, the frequency for Class 3b is generated using Equation 4 and is adjusted appropriately to address LERF sequences.

$FREQ_{class3b} = PROB_{class3b}$ x (CDF – Class 8)

= 0.00273 x (1.41E-5/yr – 6.20E-7/yr) = 3.68E-8/yr (eq. 4)

Class 4:

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

Class 5:

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

Class 6:

The Class 6 group is comprised of isolation faults that occur as a result of the accident sequence progression. For CCNPP Unit 2, this class is defined by the CCNPP Unit 2 SERF category.

$FREQ_{class6}$ = 7.96E-8/yr (eq. 5)

Class 1:

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

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

$FREQ_{class1}$ = 1.24E-5/yr – (3.70E-7/yr + 3.68E-8/yr) = 1.20E-5/yr

Class 2:

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

$FREQ_{class2}$ = 4.90E-8/yr (eq. 7)

Class 7:

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

$FREQ_{class7}$ = 9.26E-7/yr (eq. 8)

Class 8:

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

$FREQ_{class8}$ = 6.20E-7/yr (eq. 9)

Table 7 summarizes the above information by the EPRI defined classes. This table also presents dose exposures calculated using the methodology described in Reference 1. For Class 1 the person-rem is provided in Reference 7. Class 3a and 3b person-rem values are developed based on the design basis assessment of the intact containment as defined in Reference 1.

The Class 3a and 3b doses are represented as $10L_a$ and $100L_a$ respectively. Table 7 also presents the person-rem frequency data determined by multiplying the failure class frequency by the corresponding exposure.

Table 7
Baseline Risk Profile

Class	Description	Frequency (/yr)	Person-rem (calculated)[1]	Person-rem (from L_a factors)	Person-rem (/yr)
1	No containment failure	1.20E-5		3.20E+4	3.84E-1
2	Large containment isolation failures	4.90E-8	2.00E+7		9.80E-1
3a	Small isolation failures (liner breach)	3.70E-7		3.20E+5[2]	1.18E-1
3b	Large isolation failures (liner breach)	3.68E-8		3.20E+6[3]	1.18E-1
4	Small isolation failures - failure to seal (type B)	ε[1]			
5	Small isolation failures - failure to seal (type C)	ε[1]			
6	Containment isolation failures (dependent failure, personnel errors)	7.96E-8	7.01E+6		5.58E-1
7	Severe accident phenomena induced failure (early and late)	9.26E-7	5.61E+7		5.19E+1
8	Containment bypass	6.20E-7	2.25E+7		1.40E+1
	Total	1.41E-5			6.806E+1

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

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

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

Where:

$Class3a_{BASE}$ = Class 3a person-rem/year =1.18E-1 person-rem/year

$Class3b_{BASE}$ = Class 3b person-rem/year = 1.18E-1 person-rem/year

$Total_{BASE}$ = total person-rem year for baseline interval = 6.806E+1 person-rem/year (Table 7)

$\%Risk_{BASE}$ = [(1.18E-1 + 1.18E-1) / 6.806E+1] x 100 = **0.347%** (eq. 11)

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

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

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

The increase for a 10-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from 18 months to 60 months) multiplied by the existing Class 3a probability as shown in Equation 12.

$$p_{Class3a}(10y) = 0.0275 \times \left(\frac{60}{18}\right) = 0.0916 \quad \text{(eq. 12)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 13.

$$p_{Class3b}(10y) = 0.00273 \times \left(\frac{60}{18}\right) = 0.0091 \quad \text{(eq. 13)}$$

<u>Risk Impact due to 10-year Test Interval</u>

Based on the approved methodology (Reference 1) and the NEI guidance (Reference 4), the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences.

Consistent with Reference 1 the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage. Additionally the Class 1 frequency is adjusted to maintain the overall core damage frequency constant. The results of this calculation are presented in Table 8 below.

Table 8
Risk Profile for Once in Ten Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure[1]	1.11E-5	3.20E+4	3.54E-1
2	Large Containment Isolation Failures	4.90E-8	2.00E+7	9.80E-1
3a	Small Isolation Failures (Liner breach)	1.23E-6	3.20E+5	3.95E-1
3b	Large Isolation Failures (Liner breach)	1.23E-7	3.20E+6	3.93E-1
4	Small isolation failures - failure to seal (type B)	ε[3]		
5	Small isolation failures - failure to seal (type C)	ε[3]		
6	Containment Isolation Failures (dependent failure, personnel errors)	7.96E-8	7.01E+6	5.58E-1
7	Severe Accident Phenomena Induce Failure (Early and Late)	9.26E-7	5.61E+7	5.19E+1
8	Containment Bypass	6.20E-7	2.25E+7	1.40E+1
	Total	1.41E-5		6.858E+1

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

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

$\%Risk_{10} = [(Class3a_{10} + Class3b_{10}) / Total_{10}] \times 100$ (eq. 14)

Where:

$Class3a_{10}$ = Class 3a person-rem/year = 3.95E-1 person-rem/year

$Class3b_{10}$ = Class 3b person-rem/year = 3.93E-1 person-rem/year

$Total_{10}$ = total person-rem year for current 10-year interval = 6.858E+1 person-rem/year (Table 8)

$\%Risk_{10} = [(3.95E\text{-}1 + 3.93E\text{-}1) / 6.858E\text{-}1] \times 100 = \mathbf{1.149\ \%}$ (eq. 15)

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

$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100.0$ (eq. 16)

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 6.806E+1 person-rem/year (Table 7)

$Total_{10}$ = total person-rem/year for 10-year interval = 6.8588E+1 person-rem/year (Table 8)

$\Delta\%Risk_{10} = [(6.858E{+}1 - 6.806E{+}1) / 6.806E{+}1] \times 100.0 = \mathbf{0.766\%}$ (eq. 17)

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

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

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

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

A similar calculation is performed for the Class 3b probability as presented in Equation 19.

$$p_{Class3b}(15y) = 0.00273 \times \left(\frac{90}{18}\right) = 0.0137 \quad \text{(eq. 19)}$$

Risk Impact due to 15-year Test Interval

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

The increased risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage. Additionally the Class 1 frequency is adjusted to maintain the overall core damage frequency constant. The results of this calculation are presented in Table 9 below.

Table 9
Risk Profile for Once in Fifteen Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure[1]	1.04E-5	3.20E+4	3.32E-1
2	Large Containment Isolation Failures	4.90E-8	2.00E+7	9.80E-1
3a	Small Isolation Failures (Liner breach)	1.85E-6	3.20E+5	5.92E-1
3b	Large Isolation Failures (Liner breach)	1.84E-7	3.20E+6	5.89E-1
4	Small isolation failures - failure to seal (type B)	ε[3]		
5	Small isolation failures - failure to seal (type C)	ε[3]		
6	Containment Isolation Failures (dependent failure, personnel errors)	7.96E-8	7.01E+6	5.58E-1
7	Severe Accident Phenomena Induce Failure (Early and Late)	9.26E-7	5.61E+7	5.19E+1
8	Containment Bypass	6.20E-7	2.25E+7	1.40E+1
	Total	1.41E-5		6.895E+1

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

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

$\%Risk_{15} = [(Class3a_{15} + Class3b_{15}) / Total_{15}] \times 100$ (eq. 20)

Where:

$Class3a_{15}$ = Class 3a person-rem/year = 5.92E-1 person-rem/year

$Class3b_{15}$ = Class 3b person-rem/year = 5.89E-1 person-rem/year

$Total_{15}$ = total person-rem year for 15-year interval = 6.895E+1 person-rem/year (Table 13)

$\%Risk_{15}$ = [(5.92E-1 + 5.89E-1) / 6.895E+1] x 100 = **1.714%** (eq. 21)

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

$$\Delta\%Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100.0 \qquad \text{(eq. 22)}$$

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 6.806E+1 person-rem/year (Table 7)

$Total_{15}$ = total person-rem/year for 15-year interval = 6.895E+1person-rem/year (Table 9)

$$\Delta\%Risk_{15} = [(6.895E+1 - 6.806E+1) / 6.806E+1] \times 100.0 = \mathbf{1.312\%} \qquad \text{(eq. 23)}$$

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

Based on the guidance in Reference 1, the percent increase in the total integrated plant risk from a fifteen-year ILRT over a current ten-year ILRT is computed as follows:

$$\%Total_{10\text{-}15} = [(Total_{15} - Total_{10}) / Total_{10}] \times 100 \qquad \text{(eq. 24)}$$

Where:

$Total_{10}$ = total person-rem/year for 10-year interval = 6.858E+1person-rem/year (Table 8)

$Total_{15}$ = total person-rem/year for 15-year interval = 6.895E+1 person-rem/year (Table 9)

$$\%\ Total_{10\text{-}15} = [(6.895E+1 - 6.8583E+1) / 6.858E+1] \times 100 = \mathbf{0.543\%} \qquad \text{(eq. 25)}$$

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

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

From References 1, 3, 4 and 6, the Class 3a dose is assumed to be 10 times the allowable intact containment leakage, L_a (or 3,400 person-rem) and the Class 3b dose is assumed to be 100 times L_a (or 34,000 person-rem). The method for defining the dose equivalent for allowable leakage (L_a) is developed in Reference 1. This compares to a historical observed average of twice L_a. Therefore, the estimate is somewhat conservative.

Based on the EPRI method guidance (Reference 1) only Class 3 sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because for these sequences the containment remains intact. Therefore, the containment leak rate is expected to be small (less than $2L_a$). A larger leak rate would imply an impaired containment, such as Classes 2, 3, 6 and 7. Late releases are excluded regardless of the size of the leak because late releases are, by definition, not a LERF event.

Therefore, the change in the frequency of Class 3b sequences is used as the increase in LERF for CCNPP Unit 2, and the change in LERF can be determined by the differences. Reference 1 identifies that Class 3b is considered to be the contributor to LERF. Table 10 summarizes the results of the LERF evaluation that Class 3b is indicative of a LERF sequence.

Table 10
Impact on LERF due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	3.68E-8/yr	1.23E-7/yr	1.84E-7/yr
ΔLERF (3 year baseline)		8.59E-8/yr	1.47E-7/yr
ΔLERF (10 year baseline)			6.14E-8/yr

Reg. Guide 1.174 (Reference 8) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reference 1 cites Reg. Guide 1.174 and defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr. Since the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

By increasing the ILRT interval from the currently acceptable 10 years to a period of 15 years results in an increase in the contribution to LERF of 6.14E-8/yr. This value meets the guidance in Reg. Guide 1.174 defining very small changes in LERF. The LERF increase measured from the original 3-in-10-year interval to the 15-year interval is 1.47E-7/yr, which is more than the criterion presented in Regulatory Guide 1.174. The LERF increase measured from the original 3-in-10-year interval to the 15-year interval is 1.47E-7/yr, which is more than the criterion presented in Regulatory Guide 1.174. When the increase to LERF is bounded between 10-7 and 10-6 a more detailed presentation of baseline LERF is required. When the increase is less than 10-7 the change in LERF is considered allowing more flexibility with respect to the baseline LERF.

This result is not unexpected since plants with a CDF in excess of 1.0E-5/yr could have difficulty demonstrating a change in LERF less than 1.0E7/yr[9]. Since the target value is exceeded some refinement is necessary. The increase is explicitly tied to the Class 3b contribution which is generated by multiplying the total CDF by the defined split fraction (0.0027).

Using the entire CDF frequency is conservative since some sequence frequencies comprising the total CDF already account for other LERF sequences which may occur due to interfacing system LOCA events or steam generator tube ruptures. The first refinement centers on this conservatism. Sequences which result in LERF contributions are not influenced (change in outcome) by the potential for Type A leakage and can be excluded from the calculation of Class 3 leakage.

The next step to relieve conservatism examines the magnitude of the source term expected to be available for release during the accident sequence. If the debris expelled from the reactor vessel but the vessel remains essentially covered with water the source term is greatly reduced and a large source term would not be expected. Therefore if the accident sequence contains containment spray or coverage of the debris with large pools of water, the source term is not considered sufficient to support a LERF release and these contributions can be excluded. CCNPP Unit 2 PRA presents that 67.1 percent of the INTACT sequences have successful containment spray[10] therefore these sequences' frequencies can be removed.

The exclusion of INTACT scenarios where containment spray is credited and therefore scrubbing the source term release results in a frequency adjustment from 5.16E-6/yr to 1.70E-6/yr. As previously stated EPRI Class 1 consist of INTACT and LATE failures. Since LATE failures in dry containments usually infers overpressure due to steam generation where containment spray is not credited. Therefore the new Class one result is 8.95E-6/yr. This lowers the overall level 2 CDF to 1.064E-5/yr. Substituting this value into the previously defined equations and calculation method yields the results displayed in Table 11.

Table 11
Impact on LERF due to Extended Type A Testing Intervals with Adjusted CDF

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	2.28E-8/yr	7.60E-8/yr	1.14E-7/yr
ΔLERF (3 year baseline)		5.32E-8/yr	9.10E-8/yr
ΔLERF (10 year baseline)			3.80E-8/yr

The adjusted CDF yields an acceptable value (9.10E-8/yr) that meets the definition of a very small change in risk as defined in Regulatory Guide 1.174, but further refinement is achieved by analyzing the release times of the source terms.

LERF is defined by both release (large) and timing (early). It is this aspect that is examined to determine if the existing analysis can be refined further.

The characteristics release categories from the CCNPP Level 2 assessment were reviewed to examine the accident sequence progression. In particular the time at which the core is uncovered was examined. Typically, early releases involve fairly rapid voiding of the core and results in a significant quantity of radionuclides being present in the containment and available for early release.

Early release timing is typically associated with a time sufficiently short that there is an impaired ability to evacuate individuals near the plant such that a fatality could be possible. For this assessment, the breakpoint between early and late is chosen conservatively as 6.5 hours. This amount of time has been identified as sufficient for evacuation of the plant and the surrounding area.

Reviewing CCNPP's MAAP runs contained in Level 2 severe accident report[11], three cases had source terms released after the 6.5 hour mark. The first case is HRIF which simulates a loss of main feedwater due to a station blackout. The last two cases, GIOY and MRIF evaluate small LOCA's inside containment. These three MAAP cases are matched with a corresponding plant damage state (PDS) contained in CCNPP's Level 2 notebook[12]. Table 12 displays CCNPP's possible PDSs.

Table 12
Summary of Plant Damage States

PDS	Containment Bypass?	RCS Pressure at time of core damage?	Feedwater Availability?	Pressurizer PORV/SRV Status?	CHR?	AC Power Available?
1	No	High	Not available	Not stuck open	Not Available	Available
4	No	Low	Available	Not stuck open	Not Available	Available
5	No	High	Available	Not stuck open	Not Available	Available
6	No	Low	Available	Not stuck open	Available	Available
7	SGTR	N/A	N/A	N/A	N/A	N/A
8	ISLOCA	N/A	N/A	N/A	N/A	N/A
9	No	High	Available	Not stuck open	Not Available	Not Available
10	No	High	Not Available	Not stuck open	Available	Available
14	No	High	Not Available	Stuck open	Not Available	Not Available
15	No	High	Not Available	Not stuck open	Not Available	Not Available
16	No	High	Available	Stuck open	Not Available	Not Available
17	No	Low	Available	Not Stuck Open	Not Available	Not Available

The HRIF case models a loss of main feedwater as a result of a station blackout. The analysis assumes a loss of containment heat removal and ac power. The reactor coolant system is isolated and the containment is intact. Core damage occurs while the reactor coolant system is at high pressure. Based on the information in Table 12, this case can be used to represent PDS 15. Table 2-2 of Reference 12 contains the list of all the Level 1 core damage accident sequences and how each is mapped to a PDSs.

Using the correlation of the HRIF case and the SBO cases that contain a loss of feedwater (PDS 15) we can determine that the frequency contribution can be removed from the LERF contribution because the release time of the source terms is over 6.5 hours and the release would be considered late. The impacted sequences are listed in Table 13.

Table 13
SBO Sequences Removed from Total Frequency of PDS 15

Event Name	Sequence ID	Frequency (/yr)
FG0_SBO018-2	SBO018	8.25E-7
FG0_SBO010-2	SBO010	2.11E-8
FG0_SBO005-2	SBO005	1.28E-8
FG0_SBO013-2	SBO013	1.19E-8
FG0_SBO019-2	SBO019	9.68E-10
FG0_SBO015-2	SBO015	9.57E-11
FG0_SBO039-2	SBO039	7.06E-11
FG0_SBO004-2	SBO004	3.62E-12

This results in a frequency reduction of 8.72E-7/yr and a new total PDS 15 frequency of 3.10E-7/yr.

The next MAAP case evaluated is GIOY which pertains to a small LOCA inside containment with an equivalent break size of 0.005 ft^2. As described, it has an isolated containment. The reactor coolant system is at high pressure with auxiliary feedwater and ac power available. For this case, containment cooling is being provided using the containment air cooling (CAC).

Upon reviewing the MAAP results for this case the CAC maintains containment pressure. The use of CAC does not impact the release or concentration of radionuclides and the release timing is after the break point of 6.5 hours. Therefore, this case can be used to represent PDS 5. Table 14 presents the small LOCA cases assigned to this PDS which are excluded based on the late release.

Table 14
Small LOCA Sequences Removed from Total Frequency of PDS 5

Event Name	Sequence ID	Frequency (/yr)
FG0_SLOCA003-2	SLOCA003	2.84E-6
FG0_SLOCA012-2	SLOCA012	2.31E-6
FG0_SLOCA002-2	SLOCA002	5.19E-7

This results in a total frequency reduction of 5.67E-6/yr and a new total PDS 5 frequency of 7.70E-7/yr.

The last MAAP case evaluated is MRIF which pertains to a small LOCA inside containment with a break size of 0.02 ft^2. Again, the containment is isolated and the reactor coolant pressure is high. Auxiliary feedwater and containment heat removal is not unavailable. Ac power is available. Based on these characteristics, this case is associated with PDS 1. There are two small LOCA sequences that correspond to PDS 1, but the timing and impact is sufficient to address a wide range of impacts including general transient. Table 15 presents the sequences that meet the criteria for PDS 1 and their corresponding frequencies can be removed from the total frequency of PDS 1.

Table 15
General Transient and Small LOCA Sequences Removed from Total Frequency of PDS 1

Event Name	Sequence ID	Frequency (/yr)
FG0_TRAN009-2	TRAN009	9.45E-7
FG0_TRAN005-2	TRAN005	3.78E-7
FG0_TRAN008-2	TRAN008	1.66E-7
FG0_TRAN004-2	TRAN004	1.62E-8
FG0_TRAN007-2	TRAN007	6.50E-9
FG0_SLOCA007-2	SLOCA007	7.03E-10
FG0_SLOCA011-2	SLOCA011	5.42E-10
FG0_TRAN003-2	TRAN003	2.16E-10

This results in a total frequency reduction of 1.51E-6/yr and a new total PDS 1 frequency of 1.54E-6/yr. As shown in Reference 10, the exclusion of these frequencies yields new Level 2 results. Table 16 presents the new release category frequencies.

Table 16
Adjusted Release Category Frequencies

Release Category	Contributing CCNPP Unit 2 Accident Categories	Frequency (/yr)	EPRI Category
INTACT (S)	10	2.96E-6	Class 1
LERF	18	1.02E-6	Class 8
SERF	9	6.00E-8	Class 6
LATE	14	2.00E-6	Class 1
Total	n/a	6.04E-6	n/a

Substituting this value into the previously defined equations and calculation method yields the final results displayed in Table 17.

Table 17
Impact on LERF due to Extended Type A Testing Intervals with PDS Adjustments

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	6.47E-9/yr	2.16E-8/yr	3.23E-8/yr
ΔLERF (3 year baseline)		1.51E-8/yr	2.59E-8/yr
ΔLERF (10 year baseline)			1.08E-8/yr

The adjusted PDS inputs increase the margin for the LERF metric. The delta LERF between the 3 years and the 15 years is 2.59E-8/yr. This value illustrates that the proposed extension meets the definition of a very small change in risk as defined in Regulatory Guide 1.174.

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

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

$$CCFP = 1 - \left[\frac{f(ncf)}{CDF}\right] \qquad \text{(eq. 26)}$$

Where *f(ncf)* is the frequency of those sequences which result in no containment failure. This frequency is determined by summing the Class 1 and Class 3a results, and CDF is the total frequency of all core damage sequences. As previously explained it is reasonable to remove a portion of the INTACT frequency when containment spray is successful due to scrubbing. Since CCFP is only concerned with a containment failure and not whether the release is small or large, the Class 1 results without refinement must be used to calculate the CCFP.

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

$$\Delta CCFP_{10\text{-}15} = CCFP_{15} - CCFP_{10} \qquad \text{(eq. 27)}$$

Using the data previously developed the change in CCFP from the current testing interval is calculated and presented in Table 18.

Table 18
Impact on Conditional Containment Failure Probability due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	1.24E-5	1.23E-5	1.22E-5
f(ncf)/CDF	0.879	0.872	0.868
CCFP	0.121	0.128	0.132
ΔCCFP (3 year baseline)		6.09E-3	1.04E-2
ΔCCFP (10 year baseline)			4.35E-3

5.0 SENSITIVITY STUDIES

This appendix provides sensitivity studies suggested in Reference 1 for the CCNPP Unit 2 ILRT extension assessment. The evaluation includes an evaluation of assumptions made in relation to liner corrosion, the use of the expert elicitation, and the impact of external events. An additional sensitivity study looking at the large leak probability using the WCAP method presented in Reference 18.

5.1 LINER CORROSION

The analysis approach utilizes the Calvert Cliffs Nuclear Plant methodology (Reference 13) as modified by Reference 1. This methodology is an acceptable approach to incorporate the liner corrosion issue into the integrated leak rate test extension risk evaluation. The results of the analysis indicate that increasing the interval from three years to fifteen years did not significantly increase plant risk of a large early release.

Table 19 summarizes the results obtained from the CCNPP methodology utilizing plant-specific data for CCNPP Unit 2.

Table 19
CCNPP Unit 2 Liner Corrosion Risk Assessment Results Using CCNPP Methodology

<table>
<tr><th>Step</th><th>Description</th><th colspan="2">Containment Cylinder and Dome (85%)</th><th colspan="2">Containment Basemat (15%)</th></tr>
<tr><td>1</td><td>Historical liner flaw likelihood
Failure data: containment location specific
Success data: based on 70 steel-lined containments and 5.5 years since the 10 CFR 50.55a requirements of periodic visual inspections of containment surfaces</td><td colspan="2">Events 2
(Brunswick 2 and North Anna 2)
2 / (70 x 5.5) = 5.19E-03</td><td colspan="2">Events: 0
Assume a half failure
0.5 / (70 x 5.5) = 1.30E-03</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Aged adjusted liner flaw likelihood
During the 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the 5th to 10th year set to the historical failure rate.</td><td>Year
1
average 5-10
15</td><td>Failure rate
2.05E-03
5.19E-03
1.43E-02</td><td>Year
1
average 5-10
15</td><td>Failure rate
5.13E-04
1.30E-03
3.57E-03</td></tr>
<tr><td colspan="2">15 year average = 6.44E-03</td><td colspan="2">15 year average = 1.61E-03</td></tr>
<tr><td>3</td><td>Increase in flaw likelihood between 3 and 15 years
Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years.</td><td colspan="2">0.73% (1 to 3 years)
4.18% (1 to 10 years)
9.66% (1 to 15 years)</td><td colspan="2">0.18% (1 to 3 years)
1.04% (1 to 10 years)
2.41% (1 to 15 years)</td></tr>
<tr><td>4</td><td>Likelihood of breach in containment given liner flaw</td><td colspan="2">1%</td><td colspan="2">0.1%</td></tr>
</table>

Table 19 (Continued)
CCNPP Unit 2 Liner Corrosion Risk Assessment Results Using CCNPP Methodology

Step	Description	Containment Cylinder and Dome (85%)	Containment Basemat (15%)
5	Visual inspection detection failure likelihood	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 (Steps 3 x 4 x 5)	0.00073% (3 years) 0.73% x 1% x 10% 0.00418% (10 years) 4.18% x 1% x 10% 0.00966% (15 years) 9.66% x 1% x 10%	0.000180% (3 years) 0.18% x 0.1% x 100% 0.00104% (10 years) 1.04% x 0.1% x 100% 0.00241% (15 years) 2.41% x 0.1% x 100%

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

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

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

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

This likelihood is then multiplied by the non-LERF containment failures for CCNPP Unit 2. This value is calculated by the following equation for each period of interest. LERF is comprised of Class 8 and Class 3b cases (Equation 28).

Non-LERF = CDF – Class 8 – Class 3b (eq. 28)

The final adjustment to address cases with containment spray operation is used. Table 20 presents the data and the resultant increase in LERF due to liner corrosion for each case.

Table 20
Liner Corrosion LERF Adjustment Using CCNPP Methodology

Case	CDF (/yr)	Class 8 (/yr)	Class 3b (/yr)	Likelihood of Non-detected Corrosion Leakage	Increase in LERF (/yr)
3-years	4.06E-6	6.19E-7	6.47E-9	9.10E-6	3.12E-11
10-years	4.06E-6	6.19E-7	2.16E-8	5.22E-5	1.78E-10
15-years	4.06E-6	6.19E-7	3.23E-8	1.21E-4	4.12E-10

The increase in LERF per year from Table 20 is added to the Class 3b LERF cases and the sensitivity analysis performed. Table 21 provides a summary of the base case as well as the corrosion sensitivity case utilizing the refined values due successful containment spray. These values will differ from those contained in the earlier sample calculations before containment spray scenarios where removed. The "Delta Person-Rem" column provides the change in person-rem between the case without corrosion and the case that considers corrosion. Values within parentheses "( )" indicate the change or delta between the without corrosion and corrosion cases.

Table 21
CCNPP Unit 2 Summary of Base Case and Corrosion Sensitivity Cases

EPRI Class	Base Case (3 per 10 years)					1 per 10 years					1 per 15 years				
	Without Corrosion		With Corrosion			Without Corrosion		With Corrosion			Without Corrosion		With Corrosion		
	Frequency	Person-rem per year	Frequency	Person-rem per year	Delta Person-Rem per year	Frequency	Person-rem per year	Frequency	Person-rem per year	Delta Person-Rem per year	Frequency	Person-rem per year	Frequency	Person-rem per year	Delta Person-Rem per year
1	2.91E-6	9.33E-2	2.91E-6	9.33E-2	-9.99E-7	2.75E-6	8.79E-2	2.75E-6	8.79E-2	-5.71E-6	2.63E-6	8.41E-2	2.63E-6	8.41E-2	-1.32E-5
2	1.86E-8	3.72E-1	1.86E-8	3.72E-1	n/a	1.86E-8	3.72E-1	1.86E-8	3.72E-1	n/a	1.86E-8	3.72E-1	1.86E-8	3.72E-1	n/a
3a	6.50E-8	2.08E-2	6.50E-8	2.08E-2	n/a	2.17E-7	6.94E-2	2.17E-7	6.94E-2	n/a	3.25E-7	1.04E-1	3.25E-7	1.04E-1	n/a
3b	6.47E-9	2.07E-2	6.50E-9	2.08E-2	9.99E-5	2.16E-8	6.90E-2	2.17E-8	6.96E-2	5.71E-4	3.23E-8	1.03E-1	3.28E-8	1.05E-1	1.32E-3
6	6.00E-8	4.21E-1	6.00E-8	4.21E-1	n/a	6.00E-8	4.21E-1	6.00E-8	4.21E-1	n/a	6.00E-8	4.21E-1	6.00E-8	4.21E-1	n/a
7	3.73E-7	2.09E+1	3.73E-7	2.09E+1	n/a	3.73E-7	2.09E+1	3.73E-7	2.09E+1	n/a	3.73E-7	2.09E+1	3.73E-7	2.09E+1	n/a
8	6.19E-7	1.39E+1	6.19E-7	1.39E+1	n/a	6.19E-7	1.39E+1	6.19E-7	1.39E+1	n/a	6.19E-7	1.39E+1	6.19E-7	1.39E+1	n/a
CDF	4.06E-6	3.58E+1	4.06E-6	3.58E+1	9.89E-5	4.06E-6	3.59E+1	4.06E-6	3.59E+1	5.65E-4	4.06E-6	3.60E+1	4.06E-6	3.60E+1	1.31E-3
Class 3b LERF	6.47E-9		6.50E-9 (3.12E-11)			2.16E-8		2.17E-8 (1.78E-10)			3.23E-8		3.28E-8 (4.12E-10)		
Delta LERF (from base case of 3 per 10 years)						1.51E-8		1.52E-8 (1.47E-10)			2.59E-8		2.63E-8 (3.81E-10)		
Delta LERF from 1 per 10 years						n/a					1.08E-8		1.10E-8 (2.43E-10)		

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

5.2 DEFECT SENSITIVITY AND EXPERT ELICITATION SENSITIVITY

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

Using the expert knowledge, this information was extrapolated into a probability versus magnitude relationship for pre-existing containment defects. The failure mechanism analysis also used the historical ILRT data augmented with expert judgment to develop the results. Details of the expert elicitation process and results are contained in Reference 1. The expert elicitation process has the advantage of considering the available data for small leakage events, which have occurred in the data, and extrapolate those events and probabilities of occurrence to the potential for large magnitude leakage events.

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

The baseline assessment uses the Jefferys non-informative prior and the expert elicitation sensitivity study uses the results of the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency, can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the basic methodology (i.e., 10 La for small and 100 La for large) are used here. Table 22 presents the magnitudes and probabilities associated with the Jefferys non-informative prior and the expert elicitation use in the base methodology and this sensitivity case.

Table 22
CCNPP Unit 2 Summary of ILRT Extension Using Expert Elicitation Values (from Reference 1)

Leakage Size (L_a)	Jefferys Non-Informative Prior	Expert Elicitation Mean Probability of Occurrence	Percent Reduction
10	2.7E-02	3.88E-03	86%
100	2.7E-03	9.86E-04	64%

Taking the baseline analysis and using the values provided in Table 20 for the expert elicitation yields the results in Table 23 are developed.

Table 23
CCNPP Unit 2 Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
1	2.99E-06	2.97E-06	3.40E+02	2.70E-04	2.93E-06	2.50E-04	2.90E-06	2.36E-04
2	1.86E-08	1.86E-08	2.00E+07	3.72E-01	1.86E-08	3.72E-01	1.86E-08	3.72E-01
3a	N/A	1.33E-08	3.40E+03	4.54E-05	4.45E-08	1.51E-04	6.67E-08	2.27E-04
3b	N/A	3.39E-09	3.40E+04	1.15E-04	1.13E-08	3.84E-04	1.70E-08	5.76E-04
6	6.00E-08	6.00E-08	7.01E+06	4.21E-01	6.00E-08	4.21E-01	6.00E-08	4.21E-01
7	3.73E-07	3.73E-07	5.61E+07	2.09E+01	3.73E-07	2.09E+01	3.73E-07	2.09E+01
8	6.19E-07	6.19E-07	2.25E+07	1.39E+01	6.19E-07	1.39E+01	6.19E-07	1.39E+01
Totals	4.06E-06	4.06E-06	1.06E+08	3.57E+01	4.06E-06	3.57E+01	4.06E-06	3.57E+01
Δ LERF (3 per 10 yrs base)	n/a				7.91E-9		1.36E-8	
Δ LERF (1 per 10 yrs base)	n/a				n/a		5.65E-9	
CCFP	26.48%				26.68%		26.82%	

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

5.3 ESTIMATION OF RISK IMPACT FOR EXTERNAL EVENT INITIATING EVENTS

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

External events were evaluated in the CCNPP Unit 2 Individual Plant Examination of External Events (IPEEE)[14]. The IPEEE program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and an understanding of severe accident risk. The primary areas of external event analysis for the CCNPP Unit 2 IPEEE were seismic, internal fires, and other external events. All were examined but the analysis contained conservative assumptions related to consequential failures due to external events such that the absolute CDF is considered an understatement of plant performance and an over estimation of CDF.

Seismic events were addressed through a simplified seismic PRA as part of the IPEEE for CCNPP Unit 2. The Seismic PRA method screened all the components that met a high confidence low probability of failure (HCLPF) for the review level seismic event occurring with a magnitude of 0.3g. The remaining components were grouped together as a proxy component. It was assumed that if this proxy component failed it would result in core damage. This method is considered conservative.

This assessment is updated to include a consideration of improvements that have been incorporated into the internal event model. Prior seismic analyses have indicated that for a well designed plant, seismic contributions are a combination of low acceleration events with random failures and higher acceleration events with dependent component or structural failures due to forces associated with the seismic event.

As cited in NUREG-1742[15], the controlling failure typically involves prolonged loss of ac power leading to a station blackout. Low acceleration events lead to a disruption of offsite power sources and result in a prolonged need for onsite sources. This contribution has been estimated utilizing the current internal events analysis and based on the loss of offsite power (LOSP) initiating events analysis to define a conditional core damage probability (CCDP). This value is then combined with a typical estimation for the median capacity of the offsite power supply (0.3g, median capacity). The frequency is multiplied by 0.5 for the likelihood of failure of offsite sources given a seismic event.

The CCDP is calculated by modifying the @CDFALL2.CUT[16]. The modification to the cut set file sets all initiating event (IE) probabilities to 0 except for the LOSP IEs. The model contains four unique IEs that are associated with LOSP and are described as grid related, plant center, switchyard center, and weather related. The weather related LOSP IE frequency is also set to 0 instead of 1, because the weather related IE does not accurately represent a seismically induced sequence.

The remaining the IEs (IE0LOOPGR, IE0LOOPPC, and IE0LOOPSC) are modified so that their frequency is set to 1.0. This modification yields the CCDP due to LOSP. The analysis includes restoration of ac power which would not necessarily be possible for the range of seismic events. Recovery events associated with restoration of power are set to 1.0 to remove their influence on the CCDP. After modifying the probability of restoration events the SBO CCDP is 1.17E-4/yr. Reference 14 contains the Lawrence Livermore National Laboratory seismic hazard data CCNPP used in their IPEEE. From the seismic hazard curve, a 0.3g seismic event has a median frequency of 4.14E-5/yr. At this level, the probability of a loss of power is 0.5. Combining the frequency, the CCDP and the probability of offsite power yields an estimate for the frequency contribution for low acceleration seismic events. The seismic frequency estimate is 2.42E-9/yr.

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

To develop an estimate for the seismic failures for CCNPP we utilize a median capacity of 1.0g. The corresponding recurrence frequency of seismic of this acceleration or greater is 2.5E-6/yr. This is again multiplied by 0.5 to represent a median capacity. The result is 1.25E-6/yr and this is considered a bounding contribution for seismically induced failures. Combined, the CDF contribution is 1.25E-6/yr + 2.42E-9/yr from seismic initiating events.

Internal fire events have been addressed using the approached defined in NUREG-1407 and the EPRI Fire PRA Implementation Guide (FIVE method). This approach was conservative in some respects and does not reflect the current understanding with regard to fire growth and propagation. Use of this information would results in a biased assessment when compared to the current internal model which has been updated on a regular basis to provide an accurate reflection of internal risk. There is also considerable uncertainty associated with any internal fire event such that detailed results could in bias the overall conclusions without detailed presentation of these uncertainties.

The findings contained in NUREG-1742 (Reference 15) indicate that the fire CDF is primarily determined by plant transient type of events such as those from assessed plant transients. The judgment is made based on this observation that it is reasonable to assume that the ratio of intact to impaired containments will be similar for fire as for the internal events such that the total CDF and the breakdown by EPRI Class will be equivalent to that presented for the internal events.

For CCNPP Unit 2 internal events the total adjusted CDF is 4.06E-6/yr. The associated LERF contribution to Class 8 is 6.19E-7/yr and is comprised of SGTR and ISLOCA. Since SGTR and ISLOCA are unique initiators removing them from the CDF provides an approximate value for a refined internal fire assessment. Conservatively Large LOCA and Medium LOCA are retained.

CCNPP topographical location presents the opportunity for other external events. These events include tornadoes, thunderstorms, freezing precipitation, and hurricanes. Hurricanes pose approximately one threat per year and one significant threat per 10 years (Reference 15). These natural disasters would produce plant conditions similar to that of a seismic event, prolonged loss of offsite power combine with a failure of all onsite sources. The risk associated with losing both offsite and onsite power is less with these events than seismic. Therefore it is reasonable to assume the remaining external events CDF would be a magnitude of 10 less than that of the seismic CDF.

Per the guidance contained in Reference 1 the figure-of-merit for the risk impact assessment of extended ILRT intervals is given as:

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

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

$$CDF_{FIRE} = 4.06E\text{-}6/yr - 6.19E\text{-}7/yr = 3.44E\text{-}6/yr \quad \text{(eq. 29)}$$

$$CDF_{SEISMIC} = 1.25E\text{-}6/yr + 2.42E\text{-}9/yr = 1.25E\text{-}6/yr \quad \text{(eq. 30)}$$

$$CDF_{OTHER} = CDF_{SEISMIC} / 10 = 1.25E\text{-}7/yr \quad \text{(eq. 31)}$$

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

Class 3b Frequency = [(3.44E-6/yr) + (1.25E-6/yr) + (1.25E-7/yr)] * 2.7E-03 = 1.29E-8/yr

No adjustment is made to the CDF values since LERF sequences are typically associated with SGTR or interfacing system LOCA sequences which are not represented by the external event assessments. This is potentially conservative, but is reasonable based on the Due to the simplified assessment, the Given the conservative nature of the external events studies and the fact that many of the external event scenarios are long term station blackout and long term level of analysis detail. The change in LERF is estimated for the 1 in 10 year and 1 in 15 year cases and the change defined for the external events in Table 24.

Table 24
CCNPP Unit 2 Upper Bound External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 year	
External events	1.29E-8	4.31E-8	6.46E-8	5.17E-8
Internal events	6.47E-9	2.16E-8	3.23E-8	2.59E-8
Combined	1.94E-8	6.47E-8	9.70E-8	7.76E-8

The internal event results are also provided to allow a composite value to be defined. When both the internal and external event contributions are combined the total change in LERF does not exceed the guidance for very small change in risk and does not exceed the 1.0E-7/yr change in LERF. The table indicates that the external contribution is approximately twice the internal event value. The LERF increase supports the conclusion that the increased duration between tests does not result in a significant change in risk and the increase is acceptable per the criterion defined in Reference 1.

5.4 LARGE LEAK PROBABILITY SENSITIVITY STUDY

The large leak probability is a vital portion of determining the Class 3b frequency. CCNPP had previously calculated the large leak probability using the WCAP method. Table 25 present the large leak probabilities for the baseline test, 10 year test interval, and 15 year test interval[18]. Table 25 was developed using the same process as to calculate Class 3b frequency (equation number 4).

Table 25
CCNPP Unit 2 Large Leak Probabilities using the WCAP method.

Test Interval	WCAP Large Leak Probability	EPRI Accident Class 3b Frequency
3 per 10 years	2.47E-4	5.85E-10
10 years	7.41E-4	1.75E-9
15 years	1.11E-3	2.63E-9

Using the same EPRI approach, but with an updated Class 3b frequency calculated from the WCAP large leak probability data, Table 26 contains the final results.

Table 26
Impact on LERF due to Extended Type A Testing Intervals with WCAP CDF

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	5.85E-10/yr	1.75E-9/yr	2.63E-9/yr
ΔLERF (3 year baseline)		1.17E-9/yr	2.01E-09/yr
ΔLERF (10 year baseline)			8.73E-10/yr

The delta LERF values show that the EPRI methodology is conservative in predicting a large leak probability when compared to the WCAP method.

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20. San Onofre Nuclear Generating Station Living PRA, SONGS 2/3, PRA Level II Analysis Report, IPE-LEVEL2-000.

21. PRA Procedures Guide- A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, American Nuclear Society and the National Institute of Electrical and Electronic Engineers, NUREG/CR-2300, January 1983.

22. Summitt, R., Assessment of Safety Benefit for Installation of a Generator Disconnect Switch at Robinson, Ricky Summitt Consulting (RSC), Inc., RSC 98-19, June 1998.

23. Summitt, R., Comanche Peak Steam Electric Station Probabilistic Safety Assessment, Evaluation of Risk Significance of ILRT Extension, RSC, Inc., RSC 01-47/R&R-PN-110, November 2001.

24. Letter to NRC from Calvert Cliffs Nuclear Power Plant Unit No. 1: Docket No. 50-317, License Amendment Request: One-Time Integrated Leakage Rate Test Extension, dated March 27, 2002.

ATTACHMENT (4)

STATEMENT OF CALVERT CLIFFS PROBABILISTIC RISK ASSESSMENT (PRA) QUALITY

1.0 Introduction

Constellation Energy Nuclear Group (CENG) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the Probabilistic Risk Assessment (PRA) models for all operating CENG nuclear generation sites. This approach includes both 1) a proceduralized PRA maintenance and update process, and 2) the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Calvert Cliffs Nuclear Plant PRA.

2.0 PRA Maintenance and Update

The CENG risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the CENG risk management program, which consists of a CENG governing procedure and subordinate implementation procedures which:

- Delineate the responsibilities and guidelines for updating the full power internal events PRA models at all operating CENG nuclear generation sites. This includes implementing regularly scheduled and interim PRA model updates a process for tracking potential changes to the model, and processes for controlling the model and associated computer files.
- Includes the approach for performing and documenting internal PRA analysis and program updates. This includes standards for development and updates of the PRA model and supporting data.

To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following configuration control activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model. PRA screens are required for all design and procedure changes.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated based upon reviews of plant program data, particularly data supporting the Maintenance Rule.

Potential PRA model and documentation changes are captured and prioritized in the Configuration Risk Management Database (CRMP).

In addition to these activities, CENG risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.

- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for CENG nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65 (a)(4)).

3.0 Internal Events Peer Review

An independent PRA peer review was conducted under the auspices of the Pressurized Water Reactor Owners Group in June of 2010, and was performed against the requirements of Regulatory Guide 1.200, Revision 2 (Reference 1), and American Society of Mechanical Engineers (ASME)/American National Standards RA-Sa-2009 (Reference 2). The scope of the review was a full-scope review of the Calvert Cliffs Nuclear Plant (Calvert Cliffs) at-power, internal initiator PRA.

The peer review included an assessment of the PRA model maintenance and update process.

The peer review team consisted of seven members, with over 100 years of combined industry experience.

The team compared the Calvert Cliffs PRA against the applicable supporting requirement (SR) of the ASME/ANS standard. The following technical elements were assessed:

- Initiating Event Analysis (IE)
- Accident Sequence Analysis (AS)
- Success Criteria (SC)
- Systems Analysis (SY)
- Human Reliability Analysis (HR)
- Data Analysis (DA)
- Quantification (QU)
- Large Early Release Frequency (LERF) Analysis (LE)
- Internal Flood (IF), including plant partitioning, flood source identification, scenario development, initiating event analysis, and accident sequences and quantification.
- Maintenance & Update (MU)

Each SR may be assigned one or more capability categories – Cat I, Cat II, or Cat III – or may be assigned "Not Met."

In addition, facts and observations (F&O) may be assessed for SRs. These F&Os may be finding, suggestions, or best practices.

A preliminary report has been issued by the peer review team. CENG PRA personnel are reviewing the findings and have discussed them with the Peer Team lead. Based on these discussions, CENG expects the number of finding to be reassessed. In particular, the team lead concurs that many minor documentation issues assessed as "findings" could more reasonably be assessed as "observations." Also, some minor technical issues may be reassessed from "findings" to "suggestions."

The final Peer Review report is not expected to be provided until after the Calvert Cliffs Integrated Leak Rate Test (ILRT) extension request has been submitted.

The draft report identified five best practices.

The draft report identified three SRs that were "Not Met" and nine SRs that only met the requirements of "Cat I." These are described below:

SR ID	Assessment	Brief description of SR	Discussion
LE-F2	Not Met	Review LERF contributors for reasonableness.	The LERF contributors were not thoroughly reviewed for reasonableness and documented. The Calvert Cliffs PRA uses an industry standard simplified modeling approach for LERF. LERF contributors have now been reviewed, and the model was revised to address significant conservatisms. This action has now been performed for the PRA used for ILRT analysis. This will be documented in the PRA notebooks. The dominant contributors which were noted by the peer review team as conservative were interfacing system Loss Of Coolant Accidents (LOCA)s and Steam Generator Tube Rupture (SGTR). Neither of these contributors impact the results of this analysis regarding Containment integrity. Thus there is no impact to the conclusions of this analysis.

SR ID	Assessment	Brief description of SR	Discussion
LE-G5	Not Met	Identify limitations in the LERF analysis that would impact applications.	Limitations on the LERF analysis on applications were not documented. After conservatisms were addressed (see LE-F2), no significant limitations on the LERF analysis were identified. This will be documented in the PRA notebooks. No impact on ILRT analysis. See Appendix A and notes above on SR LE-F2 for a discussion on limitation on the LERF analysis.
IFQU-A10	Not Met	For each flood scenario, review LERF analysis. Modify LE analysis to account for unique flood scenarios.	No documentation identified related to the IF analysis in the LE report. After review, no modifications to LE were identified. This will be documented in the PRA notebooks. No impact on ILRT analysis.
SC-A5	Cat I met.	Specify an appropriate mission time for accident scenarios.	Some success criteria analysis did not reach a stable state at 24-hours, and the analysis did not progress past 24-hours. A review was conducted and no significant over-conservatisms were found related to curtailed mission times at 24-hours. No impact on ILRT analysis.

SR ID	Assessment	Brief description of SR	Discussion
SC-B2	Cat I met.	Restrict the use of expert judgment.	Some instance of expert judgment may be inappropriate for success criteria development of Small-Break LOCAs (SLOCA). The finding takes issue with running computer simulations of intermediate break sizes and inferring the results, instead of performing simulations to cover the whole SLOCA break spectrum. A review was conducted of this issue, and it was found that the simulations adequately represented the various break size ranges, no success criteria changes were required, and the intent of the SR is met. No impact on ILRT analysis. The PRA Notebooks will be updated to clarify and incorporate the additional reviews.
DA-B1	Cat I met.	Group components for parameter estimation.	Some groupings of component identified were not appropriate, based on service condition: salt water pumps grouped with service water pumps, and safety injection pumps grouped with auxiliary feedwater pumps. The data was updated in the PRA model causing moderate failure rate changes and this SR is addressed in the model used for the ILRT analysis. No impact on ILRT analysis.

SR ID	Assessment	Brief description of SR	Discussion
LE-C3	Cat I met.	Review significant LERF accident sequence progressions for potential repair.	No review documented for LERF accident sequences progressions that may be mitigated by repair. The Calvert Cliffs PRA uses an industry standard simplified modeling approach for LERF. The LERF model already included repair of diesel generators to recover from a station black-out. Other top sequences were subsequently reviewed and no sequences identified where a repair could be reasonably credited. No impact on ILRT analysis.
LE-C10	Cat I met.	Review significant LERF accident sequences to determine if continued equipment operation or operator actions can be supported that would reduce LERF during accident progression.	No review was documented for LERF accident sequences to identify cases where continued equipment operation or operator action can be supported that could reduce LERF during accident progression. The Calvert Cliffs PRA uses an industry standard simplified modeling approach for LERF. The model already included components that would continue to operate during accident progress, such as containment air coolers or containment spray nozzles. No instances identified where an operator action is inhibited prior to containment failure or bypass. A review of top sequences will be documented in the PRA notebooks. No impact on ILRT analysis.

SR ID	Assessment	Brief description of SR	Discussion
LE-C12	Cat I is met.	Review significant LERF accident sequences to determine if continued equipment operation or operator actions can be supported that would reduce LERF after containment failure.	No review was documented for LERF accident sequences to identify cases where LERF could be reduced after containment failure. The Calvert Cliffs PRA uses an industry standard simplified modeling approach for LERF. The model assumes that once a large early release occurs, it will be continuous. No credit is given where core damage could be arrested after containment failure. A review of top sequences did not reveal any cases where LERF could be reduced after containment failure. This will be documented in the PRA notebooks. No impact on ILRT analysis.
LE-C13	Cat I is met.	Perform a containment bypass analysis in a realistic manner.	Conservatisms were identified in the LERF analysis, particularly Interfacing System LOCA and SGTR analysis. After conservatisms were addressed (see LE-F2), no significant limitations on the LERF analysis were identified. This will be documented in the PRA notebooks. No impact on ILRT analysis.
LE-F1	Cat I is met.	Perform a quantitative evaluation of relative contribution to LERF from plant damage states and significant LERF contributors.	The evaluation of LERF contributors was not documented. The LERF contributors have been evaluated, and the model has been updated to address outliers. This will be documented in the PRA notebooks. No impact on ILRT analysis.

SR ID	Assessment	Brief description of SR	Discussion
IFEV-A6	Cat I is met.	Use various methods to determine flood initiating event frequencies.	The review of plant specific information that may affect flood likelihoods was not formally evaluated. The review will be documented in the PRA notebooks. No impact on ILRT analysis.

A report is attached at the end of this paper that discusses the impact of the peer review's technical, non-documentation findings.

4.0 CCNPP External Events PRA

Fire, Seismic, and High Wind models were incorporated into the CAFTA Fault Tree for Unit 1 prior to the incorporation of the event tree structure used in the Peer Reviewed internal events model for Calvert Cliffs. The impacts of these external events have not been fully incorporated in to the version used for the ILRT analysis. In version 4.1 of the CAFTA model the success criteria was included as top logic to mimic the success criteria macros from the previous RISKMAN Calvert Cliffs PRA model. The external event PRA models have not been peer reviewed at this point.

As noted previously the fire model used the EPRI Fire PRA Implementation Guide (FIVE method) and included a very detailed cable database of PRA modeled components. The impacts from that model are incorporated into version 4.1 of the Calvert Cliffs CAFTA Model.

The seismic model incorporated includes the impacts from the Integrated Plant Evaluation External Event model which included relatively conservative failure likelihoods for component failure likelihood due to seismic events. The current results of the Seismic model are dominated by large earthquakes which go directly to core damage and to large early release.

The high winds model used in the current CAFTA fault tree is very conservative as it assumes that all tornado strikes hit the entire site and no credit is taken for tornadoes with smaller footprints. This is there for simplicity and to give a bounding impact for tornadoes for on-line risk assessment.

The core damage frequency (CDF) for external events in version 4.1 is approximately 6E-05 and the LERF is approximately 5.5E-06. These are conservative with regards to improved event tree modeling available in version 5.2, which provides more realistic recoveries and success criteria, and updated component data which is generally a beneficial update. Note that these are Unit 1 CDF and LERF results. The Unit 2 results will be comparable and the units are similar with the exception of DG cooling and turbine trip.

5.0 Total CDF and RG 1.174

The current calculated overall CDF and LERF are currently approximately 7.5E-05 and 7.1E-06. The internal events portion is not expected to change significantly and is an accurate portrayal of risk. The fire and seismic events are generally relatively detailed but conservative modeling. The High Winds portion is very conservatively modeled as tornados are modeled as one conservative initiator.

The existing fire PRA as noted in the model is very detailed and includes a relatively conservative human action failure rate methodology. This included a very conservative human reliability analysis dependency that assumed many complete dependency failures to prevent over credit for multiple human actions versus the more detailed methods now available in the EPRI Human Reliability Analysis Calculator. Also, the existing model included an MSO evaluation and all PRA components had cables tracked. Given the updated event trees and better data for components it is expected that an update will lead to a lower or comparable fire CDF and LERF. Granted there will be some increases on specific scenarios due to revisions from NUREG/CR-6850 (Reference 3), but these will be offset by the removing the existing conservatism's noted.

The seismic PRA model is relatively conservative and other than the high magnitude acceleration event is not a dominant contributor. As with fire it will benefit from the event tree incorporation and better data in general. Thus little change is expected in seismic CDF except for a potential small reduction.

As noted the high winds model is very conservative in the tornado area in that all tornados are grouped into the most conservative event. Further, it is only a 1% contributor to external events. High winds updates are not expected to cause a significant increase in CDF or LERF. A more detailed assessment would be expected to cause a decrease in CDF.

Given the above information there is a high confidence that we are in Region II of Regulatory Guide 1.174 (Reference 4) per figures 3 and 4 for both CDF and LERF.

6.0 Other Relevant Calvert Cliffs PRA Open Items

Issues requiring action are entered into the Calvert Cliffs PRA Configuration Risk Management Program (CRMP) database as a CRMP Issue. These maintain the current list of issues where there are gaps that require closure. Issues are prioritized as to their potential impact on the calculated risk as follows:

A - Potential changes of five percent or more to CDF or LERF

B - Potential changes of one percent or more to CDF or LERF

C - Potential changes that enhance or have limited sequence impact

D - Documentation issues

The open CRMP Issues were reviewed to identify those that could have a potential impact on the proposed Unit 2 ILRT extension request. The CRMP issues which remain open and could impact this analysis are discussed below.

Open "A" CRMP Issues

CRMP ID	Description	Discussion
N/A	There are no open CRMP "A" Issues	

7.0 Conclusion Regarding PRA Capability for ILRT Extension Analysis

As described above, the CENG PRA maintenance and update processes and technical capability evaluation for the Calvert Cliffs PRA provide a robust basis for concluding that the PRA is suitable for use in this risk-informed process.

The Calvert Cliffs PRA models continue to be suitable for use in the risk informed in-service inspection application,

- The PRA maintenance and update processes in place, and
- The PRA technical capability evaluations that have been performed.

8.0 PRA Quality References

1. Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities, Revision 2, March 2009
2. ASME/ANS RA-Sa-2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008, February 2009
3. NUREG/CR-6850, EPRI/NRC – RES Fire PRA Methodology for Nuclear Power Facilities, September 2005
4. Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant Specific Changes to the Licensing Basis, Revision 1, November 2002

ENCLOSURE (1)

Technical, Non-Documentation Findings

Enclosure 1
Technical, Non-Documentation Findings

Finding F&Os

F&O Number	F&O Detail
1-16	Based on Sections 2.4 and 2.10 of the System Analysis Introduction Notebook (CO-SY-00, Rev. 0) this SR appears to be met. However, there is a potential issue related to this SR. Did not find reference to any engineering analysis needed to support Containment Air Cooler operation when this system is assumed to be available during LOSP when the containment heats up prior to electrical recovery. (This F&O originated from SR SY-B6)
Associated SR(s) AS-B3 SY-B6	Basis for Significance *Did not find reference to any engineering analysis needed to support Containment Air Cooler operation when this system is assumed to be available during LOSP when the containment heats up prior to electrical recovery.* Possible Resolution *Identify or develop engineering analysis to support PRA model.* Plant Response *An engineering analysis has subsequently been performed for the containment air cooler scenerio described in F&O 1-16. The concern would be applicable in SBO conditions where the containment heats up, and then, after power recovery, the air coolers are credited for containment pressure and temperature control. However, the analysis found that any temperature rise is not likely to challenge the capabilities of the air coolers; furthermore, the failure of the air coolers would not significantly affect risk, due to potential availability of containment spray.* Impact on Analysis *No impact on ILRT analysis. Subsequent analysis has found this issue to be non-significant: 1) the temperature rise is not likely to challenge the containment air coolers, and 2) the importance of the air coolers is significantly reduced by the redundant function provided by containment spray.*
1-17	Examined Internal Flooding Notebook (CO-IF-001, Rev. 1) Sections 3.0 and 3.1. Part of the Internal Flood analysis may not be complete for assessing the Aux Feedwater Discharge Piping as a Flood Source. (This F&O originated from SR IFSO-A1)
Associated SR(s) IFSO-A1 QU-E3	Basis for Significance *Part of the Internal Flood analysis may not be complete for assessing the Aux Feedwater Discharge Piping as a Flood Source. Table 3.1-1 contains the following statement on page 16 of CO-IF-001: "The fraction of at-power time during which the AFW system is in operation will be verified to support the final screening of the system."* Possible Resolution *Check if the review is complete and fix documentation and model, as appropriate.* Plant Response *There is no impact because analysis shows that the flooding from AFW discharge piping is not a significant contributor to CDF, and that this source may remained screened from inclusion in the model. The main reason for this is that the piping is only pressurized during testing; a small fraction of time at power. The flood contributes less than 1E-09 to CDF.* Impact on Analysis *Due to the relatively low contribution to CDF, this flood has no impact on ILRT analysis.*

Finding F&Os

F&O Number	F&O Detail
1-25	For the most part actual plant-specific data is used as a basis for the number of demands associated actual plant experiences (See basis for DA-C6), which includes both actual planned and unplanned activities. However, there are a few ESFAS testing and/or other logic channel testing that are not tracked via the plant computer. Created this F&O on non-documentation of ESFAS/logic train testing, which needs to include actual practice. (This F&O originated from SR DA-C7)
Associated SR(s) DA-C7	Basis for Significance *Non-documentation of ESFAS/logic train testing needs to include actual practice.* Possible Resolution *Document ESFAS testing and/or other logic channel testing that are not tracked via the plant computer.* Plant Response *The ESFAS logic train testing has a very low risk signficance and generally does not take the logic OOS. The train does go to 2-out-of-3 logic, which was updated. For the logic relays there is a RAW of <1.04 and Birnbaum on the order of 4E-07.* Impact on Analysis *The low risk significance of ESFAS logic train testing is considered to have no impact on ILRT analysis.*

2-3

The quantification of the model is by means of a top logic linked fault tree. This method automatically preserves the dependencies between one portion of the sequence tree and another. In the AS notebook (Ref. CO-AS-001, Rev. 1), section 3.x.7 for each event tree describes the accident sequence end states, including transfers to other event trees (e.g., ATWS, Seal LOCA, SBO, etc.).

Transfer sequences are developed for generic scenarios such as stuck open PORV or RCP seal LOCA (transfer to SLOCA ET), failure of reactor trip (transfer to ATWS ET), Station Black Out (transfer to SBO ET). However, it appears that some transfer sequences are not properly developed. The improperly developed sequences that have been identified are SGTR, VSLOCA and LOCA sequences transferring to ATWS ET as the ATWS event tree does not require any injection consistent with these initiating events.

Another issue identified is transferring from the Transient tree to the Steamline break outside containment for stuck open ADV. According to a discussion with plant PRA personal, a plant trip will most likely result in the ADV opening. The current model does not model the probability of the ADV remaining open.

(This F&O originated from SR AS-A11)

Associated SR(s)
AS-A11

Basis for Significance

It has been identified that the CCNP PRA model does not model the probability of Atmospheric Dump Valve failing to reclose at a plant trip. This is commonly referred to as consequential steamline break outside containment. A sensitivity study by the plant PRA personal showed the failure of the ADV will challenge the availability of the AFW TD pump. The CCNP PRA Staff has shown this to be a insignificant (<1%) contribution to CDF.

Possible Resolution

Model the possibility of a Steam Line Break outside containment for an ADV failing to reclose after a transient event.

Plant Response

A sensitivity case was run with the stuck open ADVs transferring to the SLBS event tree. This issue has an insignificant contribution to CDF (<1 %).

Impact on Analysis

No impact on ILRT analysis, as the issue has an insignificant contribution to CDF.

Finding F&Os

F&O Number	F&O Detail
2-9	Evidence of meeting this SR at CC-II/III is found in the PRA Data Notebook (C0-DA-001, Rev. 1) in Sections 2.1 and 2.7. Found inconsistencies in the value of total number components of different types (for both units) in Table 2-5 of the PRA Data Notebook with the actual total number for Calvert Cliffs. Also, found an inconsistency between the prior distribution and posterior distribution for SACM EDG fail to start in Table 2-6 of the Data Notebook. (This F&O originated from SR DA-D4)
Associated SR(s) DA-D4	Basis for Significance *Found an inconsistency between the prior distribution and posterior distribution for SACM EDG fail to start.* Possible Resolution *Review the inconsistent values and re-review Table 2-6 for any other inconsistencies.* Plant Response *Table 2-6 lists incorrect data and Bayesian update results for the SACMs. However, the correct values are used in the models. There is no inconsistency in the correct Bayesian update used.* Impact on Analysis *No impact on ILRT analysis. The model used for the ILRT analysis includes the correct data.*
2-10	Evidence of meeting this SR at CC-II/III is found in the PRA Data Notebook (C0-DA-001, Rev. 1) in Sections 2.1 and 2.7. Found inconsistencies in the value of total number components of different types (for both units) in Table 2-5 of the PRA Data Notebook with the actual total number for Calvert Cliffs. Also, found an inconsistency between the prior distribution and posterior distribution for SACM EDG fail to start in Table 2-6 of the Data Notebook... (This F&O originated from SR DA-D4)
Associated SR(s) DA-D4	Basis for Significance *Inconsistencies of total number of components for both units in Table 2-5 of the PRA Data Notebook with the actual number of components for a component types (125 Vdc batteries, AFW+HPSI+LPSI MDPs, and Containment Spray MDPs) in a brief scan of the table.* Possible Resolution *The table should be reviewed and data updates should be made with the correct information.* Plant Response *Data was reviewed and corrected for inconsistencies in the value of total number of components for component types. In Table 2-5, for the batteries, using four rather than eight reduces the exposure time from 701376 h to 350688 h. That increases the posterior means for type codes BA D, BA E, and BA I from 4.59E-7/h to 7.36E-7/h.* Impact on Analysis *The updated data probabilities have been included in the ILRT analysis and results in no impact on the ILRT analysis.*

Finding F&Os

F&O Number	F&O Detail
3-5 Associated SR(s) SY-A11 SY-A6	The fault tree does not include potential failures of the AFW accumulator system. (This F&O originated from SR SY-A11) Basis for Significance *AFW top gate FLWCNS is described in the AFW notebook as requiring AFW accumulators for short term operation (i.e., no instrument air is required). However, the fault tree does not include potential failures of the AFW accumulator system. This must be in the model or justified as to why it is not included.* Possible Resolution *Model components in the AFW accumulator system or justify why they do not need to be included.* Plant Response *A bounding sensitivity case was run to include failure of the AFW accumulators failing short-term AFW operation. This issue has an insignificant contribution to CDF (<1 %). Short-term failure of the AFW operation is dominated by failure of electrical support systems and failure of active hardware (i.e. valves and instrumentation).* Impact on Analysis *The addition of the air accumulators failing in the short term and included in the PRA model results in no significant increase to CDF, hence, has no impact on ILRT analysis.*
3-9 Associated SR(s) DA-B1	DA notebook table 2-5 contains the grouping of components for plant specific failure data. Many of the groupings appear to take into account differences in such things as size, type, mission type (e.g., FW TDP run vs. AFW TDP standby). However, in some cases, it is not clear what the basis for the grouping is. For example, SW MDP RUN and SRW MDP RUN are grouped together even though they are of different service conditions (salt water vs. clean water), voltages (480 VAC vs. 4160 VAC), size, etc. Similarly, AFW MDP is included with HPSI MDP and LPSI MDP, even though the two SI pumps are pumping borated water, while the AFW pump is pumping condensate grade water. No documentation of the appropriateness of these groupings is provided. (This F&O originated from SR DA-B1) Basis for Significance *In some cases, the grouping does not meet the requirements of CAT II. For example, SW MDP RUN and SRW MDP RUN are grouped together even though they are of different service conditions (salt water vs. clean water), voltages (480 VAC vs. 4160 VAC), size, etc. Similarly, AFW MDP is included with HPSI MDP and LPSI MDP, even though the two SI pumps are pumping borated water, while the AFW pump is pumping condensate grade water. No documentation of the appropriateness of these groupings is provided.* Possible Resolution *Group components by mission type and service condition and provide basis for the grouping.* Plant Response *The model has been updated to add additional component types and failure modes to better reflect service conditions. Service Water and Salt Water pumps were broken out. AFW pumps and Safety Injection pumps were broken out. This resulted in changes to the associated failure rates.* Impact on Analysis *No impact on ILRT analysis. The model used for the ILRT analysis includes the updated data and failure modes.*

Finding F&Os

F&O Number	F&O Detail
4-1	Section 2.2 of CO-SC-001 states, 'In order to be considered successful, an accident sequence must maintain a controlled stable state with the reactor subcritical, its water inventory stable, and its heat being removed. These conditions must be maintained for a mission time of 24 hours. Sequences that do not meet these criteria are binned to a core damage end state.' This statement is confusing, but the application at CALVERT CLIFFS was that any case that did not reach a stable end state in 24 hours was considered core damage. Therefore, CC I is considered met. (This F&O originated from SR SC-A5)
Associated SR(s) SC-A5	Basis for Significance *As an example, SGTR sequence 3 is success of injection and AFW but failure of isolation and RWT makeup. The PCTran case shows 27 hours before core damage, so the time available for isolation could go out that far in time. In the HRA analysis, the RWT makeup action shows the 27 hour time frame, but uses a time delay of 26 hours before the cues for RWT makeup would occur.* Possible Resolution *Run evaluations beyond 24 hours to confirm the outcome of such sequences that are classified as core damage. Credit the actual length of time available for operator actions that are significant to the CDF/LERF. Consider additional potential recoveries for sequences beyond 24 hours if they have a significant contribution to CDF/LERF. If no such sequences are significant, provide a discussion of such.* Plant Response *Reviewed top CDF/LERF cutsets to determine if modeling over conservative recoveries pass the 24 hour point. None were found. The only example where there are concems of stability at the 24 hour point is SGTR. In this case we allow recovery at the 27 hour point via the RWT refill. This timing was confirmed based on relatively conservative HPSI flow rates. This does not assume core damage at 24 hours.* Impact on Analysis *No impact on ILRT analysis, SGTR sequences do not impact intact or late failures of the Containment results. No over conservatisms where found where crediting additional recoveries beyond 24-hours would be beneficial.*

Finding F&Os

F&O Number	F&O Detail
4-5	The only mention in CO-SC-001 of shared systems between the units is the SBO EDG, noted in Section 4.1.2. It states that the SBO diesel can power any one bus on either unit. However, in the CAFTA model, there is an assumed bus preference of 11, then 24, then 12, then 23. This is noted in the EDG system notebook but no basis is provided. The procedures do not actually have a preference, which yields a potentially non-conservative analysis. For example, if there is a LOOP, the U2 diesels fail to start and the U1 diesels fail to run after 1 hour. The SBO diesel would then be aligned to U2, and it is non-conservative to give the U1 bus 11 full credit. If such non-conservatism is negligible, some analysis should be performed to demonstrate this. (This F&O originated from SR IE-A10)
Associated SR(s) IE-A10 SY-A10 IE-C3 SC-A4	Basis for Significance *Care should be taken not to credit SBO diesel at both units at the same time.* Possible Resolution *Document the basis for preferential SBO diesel assignment. Provide justification if current SBO EDG modeling is to be considered a negligible non-conservatism.* Plant Response *A bounding sensitivity case was run where the 0C DG to 4KV bus order of preference was swapped so that the order of preference favors Unit 2 first. For Unit 1, this issue has an insignificant contribution to CDF (<1%). For Unit 2, the current order of preference shows potential conservatism of <3%. Analysis identified by the system engineer suggests that the 0C diesel may be able to support more than one 4KV bus. This information may result in a model update in the future.* Impact on Analysis *No impact on ILRT analysis. The current simplified modeling for 0C DG-to-4KV bus order-of-preference has a non-significant impact or a modest conservatism.*
4-12	One basic event calculated in the appendix (ESF0HFCISZEFG) was not included in the fault tree models. CALVERT CLIFFS staff noted that it had previously been modeled, but inadvertently deleted in an update. (This F&O originated from SR HR-C1)
Associated SR(s) HR-C1	Basis for Significance *The unscreened basic events should all be included in the model.* Possible Resolution *Include the missing basic event in the fault tree model.* Plant Response *The basic will be incorporated in a future PRA model update. A sensitivity run with the basic event included the current model showed no increase in LERF is observed.* Impact on Analysis *No impact on ILRT analysis. The omission results in no measurable change in LERF and is not significant.*

Finding F&Os

F&O Number	F&O Detail
4-19	The sources of uncertainty are well identified in Table 5-1 of the LE notebook and quantified in Table 5-2 of the QU notebook. However, no discussion of the uncertainties or insights from them is provided. For example, Sensitivity 1 shows a 74% reduction in LERF, but this large reduction is not investigated. (This F&O originated from SR LE-F3)
Associated SR(s) LE-F3 LE-G4	Basis for Significance *Sensitivity analyses on the LERF results should be examined to understand the impact of the modeling assumptions and uncertain factors.* Possible Resolution *Examine the results of the LERF sensitivity analyses described in Section 5.5.2 of the LE notebook, and utilize the results of those analyses to better understand the model.* Plant Response *Dominant LERF cutsets were reviewed to identify uncertainties that could be addressed. Two changes have been implemented that addressed significant uncertainties and reduced LERF. First, a reverse-flow check valve in the CVCS Letdown line was credited as a potential ISLOCA recovery. Second, a new human action was added with realistic timing for Steam Generator isolation and RCS depressurization on a SGTR, given that Safety Injection was initially successful. These and less significant model updates resulted in a LERF-to-CDF ratio change from approximately 16 to 17% to approximately 10 to 11%. This newer ratio is in the typical range for other PWRs.* Impact on Analysis *No significant impact on ILRT analysis. The dominant LERF contributors were reviewed and model changes implemented. The Calvert Cliffs LERF contribution is now similar to other PWRs.*
4-21	The LE notebook states that limitations in the LE analysis that could impact applications is documented in the QU notebook, but it is not. Given the conservative modeling of SGTR and ISLOCA, the impact on applications should include assessment of how this conservatism can skew the LERF results. (This F&O originated from SR LE-G5)
Associated SR(s) LE-G5	Basis for Significance *Presentation of the LE limitations is important in understand how the LE model can be applied.* Possible Resolution *After performing LERF sensitivity analyses to better understand the impact of the LERF assumptions, document how the assumptions and uncertainties can impact applications.* Plant Response *Assumptions and uncertainties were reviewed to identify significant issues. After conservatisms were addressed (see discussion for F&0 4-19 above), no significant issues were identified.* Impact on Analysis *No significant impact on ILRT analysis. The dominant LERF contributors were reviewed and model changes implemented. The Calvert Cliffs LERF contribution is now similar to other PWRs.*

Finding F&Os

F&O Number	F&O Detail
4-22	The ASME PRA Standard SRs C-3, C-10 and C-13 require a review of the LERF results for conservatism in the following areas: 1. Engineering analyses to support continued equipment operation or operator actions during severe accident progression that could reduce the LERF. 2. Engineering analyses to support continued equipment operation or operations after containment failure. 3. Potential credit for repair of equipment. No such review has been performed, despite the large conservatism noted in the containment bypasses. (This F&O originated from SR LE-C10)
Associated SR(s) LE-C10 LE-C12 LE-F2 LE-C3	Basis for Significance *Excessive conservatism can skew the LERF results, and the ASME standard calls for a results review to consider each of these items.* Possible Resolution *Review results for excessive conservatism as indicated in the SRs.* Plant Response *The LERF results were reviewed for conservatisms as described in the SRs. After conservatisms were addressed (see discussion for F&0 4-19 above), no significant issues were identified.* Impact on Analysis *No significant impact on ILRT analysis. The dominant LERF contributors were reviewed and model changes implemented. The Calvert Cliffs LERF contribution is now similar to other PWRs.*

Finding F&Os

F&O Number	F&O Detail
4-23	Conservatisms in the ISLOCA analyses were discussed in the AS review. SGTR was treated in an overly conservative manner by categorizing all SGTR as LERF. (This F&O originated from SR LE-C13)
Associated SR(s) LE-C13	Basis for Significance *The over conservatism in the ISLOCA and SGTR analyses dominates the LERF results.* Possible Resolution *Perform more realistic analyses on SGTR and ISLOCA (or at least perform sensitivities to understand the impact of the conservatism).* Plant Response *The LERF results were reviewed for conservatisms as described in the SRs. After conservatisms were addressed for SGTR and ISLOCA (see discussion for F&0 4-19 above), no significant issues were identified.* Impact on Analysis *No significant impact on ILRT analysis. The dominant LERF contributors were reviewed and model changes implemented. The Calvert Cliffs LERF contribution is now similar to other PWRs.*
4-24	The LERF contributors have not been reviewed for reasonableness. The QU notebook discusses the top 20 LERF cutsets (which total 73% of the total LERF). It notes conservatism in the cutsets and says it will be evaluated in Section 5.2, but is not. Section 4.3.6 of the QU notebook compares the total LERF of CALVERT CLIFFS to St. Lucie, but does not even break the results down by contributor (e.g., SGTR, ISLOCA, etc.) (This F&O originated from SR LE-F2)
Associated SR(s) LE-F2	Basis for Significance *The SR requires a review of the contributors for reasonableness.* Possible Resolution *Review the significant LERF contributors for reasonableness, and if needed, modify the model to make the LERF more realistic.* Plant Response *The LERF results were reviewed for reasonableness. After conservatisms were addressed (see discussion for F&0 4-19 above), no significant issues were identified.* Impact on Analysis *No significant impact on ILRT analysis. The dominant LERF contributors were reviewed and model changes implemented. The Calvert Cliffs LERF contribution is now similar to other PWRs.*

Finding F&Os

F&O Number	F&O Detail
5-10	Following the failure of one or more containment penetrations to isolate on CIAS, a feasible operator action is to manually close the failed valves from the Main Control Room. (This F&O originated from SR LE-D7)
Associated SR(s) LE-D7	Basis for Significance *Containment Isolation failure probability is not negligible. It is on the order of 1E-03. Level 1 cutsets with frequencies at the 1E-6 and 1E-7 levels would result in LERF cutsets at the 1E-9 and 1E-10 levels.* *By adding a recovery of failed containment isolation valves, the LERF results may be reduced to their more realistic estimate.* Possible Resolution *Consider the merits from adding an operator action to close the penetrations valve(s) that should have isolated on CIAS but did not. If creditable, develop an HRA for that action.* Plant Response *The merits have been considered of adding an operator action in order close containment penetration in order to recover from a containment isolation failure. Containment isolation contributes a small, but significant amount to overall LERF. However, a review of top cutsets shows that a recovery is not feasible for those top sequences, because the sequence includes either 1) a loss of CR indication, 2) includes a station black-out condition, or 3) includes non-recoverable pipe breaks.* Impact on Analysis *The impact on ILRT analysis is not significant, as containment isolation recovery is not plausible for the significant sequences where containment isolation has failed.*

Finding F&Os

F&O Number	F&O Detail
5-17	Bayesian updates of non-time-based LOCA data were improper. The small and medium LOCA frequencies were obtained from draft NUREG 1829 then Bayesian updated (in App E) with CALVERT CLIFFS experience from 2004 to 2008. The Very Small LOCA prior having alpha = 0.4, Mean = 1.57E-03; was Bayesian updated to a Posterior having a mean value of 7.02E-04. This represents an excessive drop associated with CALVERT CLIFFS experience of 4 to 5 years. Similarly, the Small and Medium LOCAs were Bayesian updated with the whole industry experience rcy data. The draft NUREG 1829 LOCA frequencies were obtained from expert elicitations (not time-based) that included crack propagation analysis. The Bayesian update for VSLOCA used the Alpha parameter and the mean value to justify that the prior mean was based on 255 rcy. This may not have been the basis for the expert elicitations in NUREG 1829. (This F&O originated from SR IE-C1)
Associated SR(s) IE-C1 IE-C4	Basis for Significance *The Industry data over which the Bayesian updating is performed should be based on a point estimate relating the industry events over the total number of industry PWR reactor-critical-years (such as SLOCA frequency in NUREG/CR 5750). In the case of VSLOCA, SLOCA, and MLOCA, the NUREG/ 6928 values were obtained through expert elicitation (not time-based). Therefore, the number of PWR rcy may not have been a main factor.* Possible Resolution *Either perform no Bayesian update on the NUREG/CR 6928 LOCA frequencies or use the NUREG/CR 5750 time-based data as a prior.* Plant Response

Finding F&Os

F&O Number	F&O Detail
	CENG understands the general concern on Bayesian updating of rare events. However, the method used was based on a paper of data experts regarding LOCA frequencies. These experts included INL, NRC and Industry experts. In addition, the approach used for the Calvert PRA was the same as used for the NRC SPAR model. *The approach used in the CALVERT CLIFFS PRA was to use plant-specific LOCA size intervals combined with exceedance frequencies from NUREG-1829 to obtain LOCA frequencies. Then the VSLOCA, SLOCA, and MLOCA frequencies were modified with a Bayesian update using industry experience over 2004 – 2008 to account for no such LOCAs since the expert elicitation had occurred. This approach was developed at the Idaho National Laboratory (INL) to generate LOCA frequencies for the standardized plant analysis risk (SPAR) models maintained for the U.S. Nuclear Regulatory Commission (NRC). The methodology and results are documented in a peer-reviewed paper presented at the PSA'08 Conference. One of the co-authors was an NRC employee and another was a highly-respected statistician. Therefore, this approach is deemed an acceptable approach to determining LOCA frequencies for U.S. pressurized water reactors (PWRs) for use in probabilistic risk assessment (PRA) models.* *NUREG-1829 is the most recent source for determining LOCA frequencies for U.S. commercial nuclear power plants. However, the expert elicitation process conducted in 2003 and documented in that report increased the small and medium LOCA (SLOCA and MLOCA) frequencies to account for pressurized water stress corrosion cracking (PWSCC) concerns. The results from that report are therefore higher than those that would be generated using only historical data. For example, using historical data from 1988 – 2007, there were no SLOCAs (0.5 to 2.0 in. equivalent diameter) over 1179 reactor critical years (rcry). Using a Jeffreys non-informative gamma prior (0.5, 0) and a Bayesian update, the historical evidence results in a SLOCA mean frequency of 4.24E-4/rcry (0.5 divided by 1179 rcry). However, using NUREG-1829, the resulting mean frequency is 1.96E-3/rcry (higher because of PWSCC concerns). That document also indicated median and 95th percentiles for its exceedance frequencies. Using that information, the gamma distribution alpha factor is approximately 0.4 for the SLOCA distribution. This distribution implies that one SLOCA is expected by the experts used in NUREG-1829 within the U.S. PWRs every 510 rcry (inverse of 1.96E-3/rcry), starting with 2004. Given approximately 63 PWR rcry each year, this implies a SLOCA approximately every 8 years.* *Because the NUREG-1829 authors indicate that they expect PWSCC concerns to be mitigated in the future, the dilemma is whether to use the higher PWSCC- influenced SLOCA frequency from NUREG-1829 until that document is updated, which probably will not occur for many years. One alternative is to use U.S. PWR experience since 2003 (the end year for the expert elicitation process in NUREG-1829) in a Bayesian update of the NUREG-1829 frequency distribution. That approach is what is described in the PSA'08 paper and was used in the CALVERT CLIFFS PRA update. This approach allows one to periodically update the NUREG-1829 frequency based on the PWR experience. Results show a smooth transition from the higher PWSCC-influenced frequencies from NUREG-1829 to the historical evidence frequencies as time since 2003 progresses. In all cases, the Bayesian updated result lies between the NUREG-1829 and historical evidence results.* *Note that the CALVERT CLIFFS results are similar but not exactly the same because of plant-specific LOCA size intervals.* Impact on Analysis *No impact on ILRT analysis. The approach used for LOCA frequencies has been validated by industry experts and is the same approach as was used for the NRC's SPAR model.*

Finding F&Os

F&O Number	F&O Detail
5-19	The Medium LOCA frequency may be classified as extremely rare event. It would require no Bayesian updating. The CENG current CCNP SLOCA and MLOCA frequencies are very close even though the source data in NUREG 1829 indicates a negative exponential drop in these frequencies. (This F&O originated from SR IE-C13)
Associated SR(s) IE-C13	Basis for Significance *No statistically significant plant experience that warrants Bayesian updating the MLOCA frequency.* Possible Resolution *Do not Bayesian update IE MLOCA.* Plant Response *There is no rule stating that plant-specific evidence cannot be used to update rare event frequencies. The ASME/ANS standard recommends that such Bayesian updates not be used for such rare events. This recommendation was made because of the large impact a single rare event occurring at a plant could have on such an update process. However, CALVERT CLIFFS has not experienced any rare event occurrences, so this concern does not come into play. In addition, the limited CALVERT CLIFFS experience results in only small changes to the rare event prior distributions. See response to 5-17 above.* Impact on Analysis *The small changes that results from the limited CALVERT CLIFFS experience has no impact on ILRT analysis.*
5-23	The Pre-Initiator HRAs did not include the miscalibration of SIT pressure. For example, in the event where SIT pressure is miscalibrated high, various accident scenarios requiring SI are negatively impacted. Add SIT pressure miscalibrated high or, justify no impact on CDF / LERF. (This F&O originated from SR HR-A2)
Associated SR(s) HR-A2	Basis for Significance *This event may impact event timings and expected plant responses and requires more investigation.* Possible Resolution *Add this pre-initiator. Or, justify its insignificance and this justification to the HR notebook.* Plant Response *It is agreed that the miscalibration of SIT pressure could have a negative impact on various accident scenarios involving LLOCA and VLLOCA initiators. However, this instrumentation is not modeled explicitly and is therefore deemed included within the component boundary for the SIT. As such the miscalibration probability would be included in the SIT unavailability. The explicit modeling of this type of miscalibration would set a precedent for modeling miscalibration for all instrumentation as well as equipment protection signals, with probably no discernable benefit.* Impact on Analysis *Given the pressure of the CALVERT CLIFFS SITs they are only required and provide significant benefit on Large LOCAs. The frequency of a Large LOCA times the pre-initiator frequency is negligible for impact, hence has no impact on ILRT analysis.*

Finding F&Os

F&O Number	F&O Detail
5-30	Section 3.2.11 discussed the containment challenge from Hydrogen Combustion. It concluded that the challenge may be significant for some accident scenarios. The CCNP entry in Table 6.11-2 of the Level 2 WCAP showed a potentially significant impact from Hydrogen burn. Provide an estimate of the impact of Hydrogen burn on containment pressure. Use an accident scenario that is likely to produce larger amounts of H2 with failed containment spray. The optimal time to estimate the impact of Hydrogen burn is approximately at 2 hours which is the time when the EOF and TSC personnel have convened and are ready to guide the Main Control Room into periodic Hydrogen burns before the formation of explosive mixtures. (This F&O originated from SR LE-D1)
Associated SR(s) LE-D1 LE-B2	Basis for Significance *Explosive Hydrogen mixture is one of the challenges to containment structure.* Possible Resolution *Update the LE notebook with the new estimate of Hydrogen bum at 2 hours impact on containment integrity.* Plant Response *The Calvert Cliffs PRA Level 2 analysis relied on the "Simplified Level 2 Modeling Guidelines" established in Westinghouse document WCAP-163341-P. This document is also the basis for the Level 2 analysis for many other PWRs. Explosive Hydrogen combustion is considered in the Level 2 analysis, and in some sequences it is the dominant contributor to containment failure. Furthermore, Westinghouse personnel were contacted and asked if there were other cited concems conceming the hydrogen bum analysis in the WCAP, none were noted. The WCAP methodology provides a reasonable evaluation of hydrogen bum.* Impact on Analysis *No impact on ILTR analysis. The industry-standard methodology provides a reasonable evaluation of hydrogen bum.*

Finding F&Os

F&O Number	F&O Detail
6-3	Expert judgment was not used as the sole basis for any success criteria. However, upon inspection of the PCTran run tables in the SC report appendices, many instances of surrogate or inferred results were found. Instead of running specific PCTran calculations to cover the whole SLOCA break size spectrum, intermediate break sizes have been calculated supplemented with expert judgment to derive limiting time delay for operators to actuate SI (30 min) or limiting time delay for OTCC (SGL<350'+10min). (This F&O originated from SR SC-B2)
Associated SR(s) SC-B2	Basis for Significance *Instead of running specific PCTran calculations to cover the whole SLOCA break size spectrum, intermediate break sizes have been calculated supplemented with expert judgment to derive limiting time delay for operator actions. Using surrogate or inferred results is seen as using expert judgment to assume results rather than documenting the individual cases with PCTran (which is cited in the CALVERT CLIFFS PRA for its ease of use as well as speed, making various scenario evaluations easy to complete). There are numerous PCTran cases where this occurs.* Possible Resolution *Either run and fully document the actual cases specified in the SC appendices or indicate throughout the SC report where inferred results are used (and why doing so is appropriate) as well as which PCTran cases are fully documented as complete scenarios. Additionally, better documentation of the basis for all of the PCTran scenarios throughout the PCTran scenario documentation report is needed as there is very little indication as to why the various scenarios were run in the way they were or why additional runs were not required.* Plant Response *The approach for SLOCA break size analysis is discussed in the Success Criteria notebook. Furthermore, a review was conducted of this issue, and it was found that the computer simulations adequately represented the various break-size ranges. If more runs were added, as inferred by this potential finding, the result would be marginally more information on timing for recovery of auto start of safety injection equipment. As conservative timing is used for this event (to minimize number of combinations of dependent HRA events) and this action was not found to have a large impact, there would be few insights gained by large amounts of documentation for additional runs. The success criteria from a safety injection standpoint is the same for the LOCAs in question.* Impact on Analysis *The existing analysis meets the intent of the SR and therefore there is no impact on ILRT analysis.*

Finding F&Os

F&O Number	F&O Detail
6-5	When appropriate, the simultaneous unavailability within a system is documented in the system notebooks and included in the PRA model. However, a further review of these items is required for completeness. (This F&O originated from SR SY-A20)
Associated SR(s) SY-A20	Basis for Significance *While the intent of this SR is met and generally well documented and implemented, some instances of multiple equipment unavailabilities were found to not be included in the PRA model at all places required for all of the indicated equipment failures contained in the system notebook. As an example, in the AFW notebook, event AFW0TMMAINT6-F7 is related to pumps 12 and 13 being unavailable, but is only modeled as an event at pump 13 in the PRA model with pump 12 having no common unavailability event. Additionally, event AFW0TMMAINT-TF is modeled as an unavailability of pumps 11 and 12 and is modeled as an event at both pumps, but is not listed in the AFW system notebook. Other similar items may exist and need to be corrected.* Possible Resolution *Perform a review of all systems for simultaneous unavailability events in the PRA model and in the various system notebooks to ensure consistency. Correct any identified missing events or event descriptions in the PRA model and system notebooks.* Plant Response *AFW basic event AFW0TMMAINT6-F7 was determined to not be needed in the plant model. The basic event was removed and a sensitivity run was performed. There was no significant change in CDF/LERF. All remaining AFW equipment unavailability events in the model and notebooks were reviewed for consistency. AFW0TMMAINT-TF was determined to be modeled correctly. Its description was found to be in error in the system notebook. A review for concurrent maintenance was previously performed and documented in the Data Notebook.* Impact on Analysis *The removal of basic event AFW0TMMAINT6-F7 was not significant and has no impact on ILRT analysis.*
6-22	Upon RAS, LPSI stops and EOP-5, Step S.1(d) requires the Operators to 'Shut RWT OUT Valves SI-4142, 4143'. This manual action was not modeled in the PRA. The CALVERT CLIFFS PRA staff provided reasonable response to this issue. Based on CR-2009-005581, there is no impact on pump operability. Also, the staff will continue to track the CR. If there are any changes to the disposition of pump operability, then a new HRA may be added to the PRA model (if warranted). (This F&O originated from SR HR-E1)
Associated SR(s) HR-E1	Basis for Significance *The time window between RAS and RWT level lower than NPSH requirement for LPSI is likely to be few minutes only.* Possible Resolution *Justify that no HRA is required. Or add this new HRA to the PRA model.* Plant Response *As discussed during the Peer Review, shutting of the RWT outlet valves upon a RAS does not impact station operability concerns and is documented in CR-2009-005881. Currently there are no calculations saying that the SI and CS pumps will fail if the RWT isolation valves are not closed within a certain time period after RAS. Due to potential vortexing issues, Design Engineering is investigating having these valves shut automatically upon a RAS to avoid these issues. If the ongoing investigation determines that the pump disposition is changed anew HRA will be added to the PRA model.* Impact on Analysis The operability of the system as documented above shows there is no impact on CDF, hence, no impact on ILRT analysis.

Finding F&Os

F&O Number	F&O Detail
6-23 Associated SR(s) HR-G7	When the Calculator reads in the combinations, it assumes that actions occur in the order of the time delay (Td). However, the time delay is not the same for all sequences, and care must be taken to make the combinations appropriate for the sequences in which they occur. Page 88 of the HRA notebook indicates this was considered, since the Td was modified for events occurring prior to reactor trip, and also for OTCC after SG overfill. However, not all occurrences have been addressed. The combination examined by the review team is Combination 770 (OTCC after CST depletion). In this event the CST depletion should come first. (This F&O originated from SR HR-G7) Basis for Significance *If CST depletion occurs first, then there is a shorter time between the actions (CALVERT CLIFFS estimate is 5 or 6 hours), which is not sufficient to credit shift turnover. Processing this through the dependency event tree should yield a Low Dependence (since stress level for OTCC was assessed as high in Table 3-22). This would raise the joint HEP of this combination by a couple orders of magnitude, impacting the results. There may be other similar items.* Possible Resolution *Correct the evaluation of the combination presented. Review (and document to the extent reasonable) the consideration of sequential timing of actions in each combination assessment.* Plant Response *The human action dependency evaluation has been updated to account for appropriate sequencing of events. This has been incorporated into the plant model.* Impact on Analysis *The PRA model used for the ILRT analysis incorporated the update human action dependency evaluation, and therefore there is no impact on the ILRT analysis.*
7-7 Associated SR(s) AS-A11	Transfer sequences are developed for generic scenarios such as stuck open PORV or RCP seal LOCA (transfer to SLOCA ET), failure of reactor trip (transfer to ATWS ET), Station Black Out (transfer to SBO ET). However, it appears that SGTR, VSLOCA and LOCA sequences transferring to ATWS ET are not properly modeled as the ATWS event tree does not require any injection consistent with these initiating events. (This F&O originated from SR AS-A11) Basis for Significance *LOCA (or SGTR) sequences associated to the failure of reactor trip are transferred to a "generic" ATWS event tree. There is a need to control primary mass inventory which is lost when transferring these sequences to the "generic" ATWS ET.* Possible Resolution *As these transferred sequences are most probably very low contributors to CDF and LERF, they can simply be associated to Core Damage. Or, further documentation is needed to justify adequate modeling.* Plant Response *The transfer sequence to the ATWS event tree is adequate. The lack of a inventory control top event is not significant, and would change CDF by less than .001%.* Impact on Analysis *The lack of inventory control in the ATWS event tree is not significant, and would therefore not impact the ILRT analysis.*