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10 CFR 50.90

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Subject: Request for Amendment to Technical Specification 5.5.16, "Containment Leakage Rate Testing Program"

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) hereby requests an amendment to Appendix A, Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station Units 1 and 2, respectively.

Specifically, the proposed changes will revise TS 5.5.16, "Containment Leakage Rate Testing Program," to reflect a one-time five-year deferral of the containment Type A, integrated leak rate test from once in ten years to once in 15 years.

TS Section 5.5.16 establishes the program for leakage rate testing of the containments, as required by 10 CFR 50.54, "Conditions of licenses," Section (o) and 10 CFR 50, Appendix J, Option B, "Performance Based Requirements," as modified by approved exemptions. Additionally, the testing conforms to the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

EGC has assessed the risk implications of extending the Braidwood Station and Byron Station Type A test interval from once in ten years to once in 15 years. This evaluation indicated that the analyzed Integrated Leak Rate Test (ILRT) interval extension has a minimal impact on public risk. In addition, the extension of the ILRT test interval is consistent with extensions recently granted to other licensees.

The information supporting the proposed TS changes is subdivided as follows.

Attachment 1 provides an evaluation of the proposed changes.

Attachment 2A contains the copy of the marked up TS page for Braidwood Station.

Attachment 2B contains the copy of the marked up TS page for Byron Station.

Attachment 3A provides the retyped TS page for Braidwood Station.

Attachment 3B provides the retyped TS page for Byron Station

Attachment 4 provides the risk assessment supporting the proposed change for Braidwood Station

Attachment 5 provides the risk assessment supporting the proposed change for Byron Station

There are no regulatory commitments in this License Amendment Request.

The proposed TS changes have been reviewed by the Braidwood Station and Byron Station Plant Operations Review Committees (PORCs) and approved by the Nuclear Safety Review Board (NSRB) in accordance with the EGC Quality Assurance Program.

EGC is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated State Official.

We request approval of the proposed changes by February 19, 2008 with an implementation period of 30 days.

Should you have any questions concerning this submittal, please contact Mr. John L. Schrage at (630) 657-2821.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 4th day of April 2007.

Sincerely,

Darin M. Benyak
Manager - Licensing
Exelon Generation Company, LLC

Attachments:

Attachment 1	Evaluation of Proposed Changes
Attachment 2A	Markup of Proposed Technical Specification Page, Braidwood Station
Attachment 2B	Markup of Proposed Technical Specification Page, Byron Station
Attachment 3A	Retyped Technical Specification Page, Braidwood Station
Attachment 3B	Retyped Technical Specification Page, Byron Station
Attachment 4	Risk Assessment for Braidwood Unit 1 and Unit 2 to Support ILRT (Type A) Interval Extension Request
Attachment 5	Risk Assessment for Braidwood Unit 1 and Unit 2 to Support ILRT (Type A) Interval Extension Request

cc: Regional Administrator - Region III, NRC
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station

ATTACHMENT 1
Evaluation of Proposed Changes

1.0 INTRODUCTION

2.0 DESCRIPTION OF PROPOSED AMENDMENT

3.0 BACKGROUND

4.0 TECHNICAL ANALYSIS

5.0 REGULATORY ANALYSIS

- 5.1 No Significant Hazards Consideration
- 5.2 Applicable Regulatory Requirements/Criteria

6.0 ENVIRONMENTAL CONSIDERATION

7.0 PRECEDENT

8.0 REFERENCES

ATTACHMENT 1
Evaluation of Proposed Changes

1.0 INTRODUCTION

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC, (EGC) hereby requests the following amendment to Appendix A, Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station Units 1 and 2, respectively. Specifically, the proposed changes will revise TS 5.5.16, "Containment Leakage Rate Testing Program," to reflect a one-time, five-year extension of the current containment Type A test interval requirement.

EGC is requesting this one-time amendment in anticipation of a rule change to 10 CFR 50 extending the Type A testing frequency to at least 15 years. Approval of the proposed change will allow sufficient time for this rule change to be processed and incorporated into the Braidwood Station and Byron Station TSs. The proposed change is consistent with Integrated Leak Rate Test (ILRT) extensions recently granted to other licensees (i.e., as listed in Section 7.0, "Precedent").

Braidwood and Byron Station TS Section 5.5.16 establishes the leakage rate testing requirement for the containment, as required by 10 CFR 50.54, "Conditions of licenses," Section (o) and 10 CFR 50, Appendix J, Option B, "Performance Based Requirements," and modified by approved exemptions. Additionally, the testing conforms to the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The proposed change will allow the Type A test to be performed within 15 years of the most recent Type A test at Braidwood Station and Byron Station, as described below.

Braidwood Station Unit 1:	No later than October 5, 2013
Braidwood Station Unit 2:	No later than May 4, 2014
Byron Station Unit 1:	No later than February 19, 2013
Byron Station Unit 2:	No later than November 2, 2014

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed change adds two new exceptions to TS 5.5.16 that modify the schedule for the next Type A test for Units 1 and 2 at Braidwood Station and Byron Station, to a 15-year interval. The proposed wording associated with the exceptions to be added to TS 5.5.16 is identified below in bold type.

2.1 Braidwood Station Technical Specifications

"5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94 01, Revision 0, **as modified by the following exceptions:**

1. **NEI 94-01 - 1995, Section 9.2.3: The first Unit 1 Type A test performed after October 5, 1998 Type A test shall be performed no later than October 5, 2013.**
2. **NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after May 4, 1999 Type A test shall be performed no later than May 4, 2014."**

2.2 Byron Station Technical Specifications

"5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94 01, Revision 0, **as modified by the following exceptions:**

1. **NEI 94-01 - 1995, Section 9.2.3: The first Unit 1 Type A test performed after February 19, 1998 Type A test shall be performed no later than February 19, 2013.**
2. **NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after November 2, 1999 Type A test shall be performed no later than November 2, 2014."**

3.0 BACKGROUND

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

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. The cylinder wall is pre-stressed with a post-tensioning system in the vertical and horizontal directions, and the dome roof is pre-stressed utilizing a three way post-tensioning system.

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

The integrity of the containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs), and the overall leak tight integrity of the containment is verified by a Type A ILRT, as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak tight characteristics of the containment at the design basis accident pressure.

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

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

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

RG 1.163, Section C, "Regulatory Position," Exception 3 discusses the visual examination of accessible internal and exterior surfaces of the containment system for structural problems. Exception 3 further states, "These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the

Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration."

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

With the two most recent successful Type A tests at both Braidwood Station and Byron Station, and greater than 24 months of elapsed time between the two tests, both stations currently have a test interval of once every 10 years. The current 10-year interval for the completion of the next Type A test for each unit at both stations ends on the following dates:

Braidwood Station Unit 1:	October 5, 2008
Braidwood Station Unit 2:	May 4, 2009
Byron Station Unit 1:	February 19, 2008
Byron Station Unit 2:	November 2, 2009

The Braidwood Station and Byron Station containment leakage testing program complies with the requirements of the General Design Criteria and Appendix J of 10 CFR 50. The Type A test is performed at a frequency of once every 10 years, based on Type A test performance history.

The maximum allowable leakage rate, L_a at pressure P_a, is 0.075 weight percent (%) per day for the full pressure test. The Type A test is performed in accordance with the provisions of ANSI N56.8-1994.

Visual inspection of the accessible interior and exterior surfaces of the containment structures and components is performed in accordance with the requirements of 10 CFR 50 Appendix J, RG 1.163, and NEI 94-01 to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness. If there is evidence of structural deterioration, a Type A test is not performed until all identified irregularities are resolved, in accordance with acceptable procedures, nondestructive tests, and inspections.

4.0 TECHNICAL ANALYSIS

The testing requirements of 10 CFR 50, Appendix J provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed allowable leakage rate values specified in the TS and Bases. The allowable leakage rate is limited such that the leakage assumptions in the safety analyses are not exceeded. The limitation of containment leakage provides assurance that the containment would perform its design function following an accident, up to and including the design basis accident.

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under Option A "Prescriptive Requirements" or Option B. The NRC issued License Amendments Nos. 73 and 81 on April 4, 1996 for Braidwood Station and Byron Station respectively, to permit implementation of 10 CFR 50, Appendix J, Option B. TS 5.5.16 currently requires the establishment of a Containment Leakage Rate Testing Program in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program implements the guidelines contained in RG 1.163, which

specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01, subject to several regulatory positions stated in RG. 1.163.

10 CFR 50, Appendix J, Section V, "Inspection and Reporting of Tests," Option B specifies that the regulatory guide (i.e., RG 1.163) or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TSs. Additionally, deviations from guidelines endorsed in the Regulatory Guide are to be submitted as a revision to the plant's TS. Therefore, this application does not require an exemption from 10CFR 50, Appendix J, Option B.

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

Further justification for the proposed change is based on research documented in NUREG-1493, "Performance-Based Containment Leak-Test Program." NUREG-1493 made the following observation with regard to changing the test frequency:

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

4.1 Summary of Test and Inspection Programs

Satisfactory results from previous Type A tests at both Braidwood Station and Byron Station, as well as continued satisfactory results from Type B and Type C Local Leak Rate Tests and containment inspections support the proposed one-time extension of the containment Type A test interval.

4.1.1 Integrated Leak Rate Test History

Type A testing is performed to verify the integrity of the containment structure in its LOCA configuration. As stated in NEI 94-01, "The purpose of Type A testing is to verify the leakage integrity of the containment structure. The primary performance objective of the Type A test is not to quantify the overall containment system leakage rate." The Type A testing methodology that is described in ANSI/ANS 56.8-1994, and the modified testing frequencies that are recommended by NEI 94-01 serve to ensure continued leakage integrity of the containment structure. The results of the previous two ILRTs for each containment structure at Braidwood Station and Byron Station, which are presented in Tables 4.1.1.a through 4.1.1.d, validate the leakage integrity of the containment structures.

These historical results indicate that the Braidwood Station and Byron Station containment structures meet the requirements of 10 CFR 50 Appendix J, thus ensuring essentially a leak tight barrier. These plant specific results support the conclusions of NUREG-1493.

Table 4.1.1a
Braidwood Station Unit 1 ILRT Test Results
(% weight per day)

ILRT Test Date	Total Leakage (Note)	Acceptance Limit (Note)
11/95	0.064%	< 0.075%
10/98	0.071%	< 0.075%

Table 4.1.1b
Braidwood Station Unit 2 ILRT Test Results
(% weight per day)

ILRT Test Date	Total Leakage (Note)	Acceptance Limit (Note)
11/94	0.053%	< 0.075%
5/99	0.063%	< 0.075%

Table 4.1.1c
Byron Station Unit 1 ILRT Test Results
(% weight per day)

ILRT Test Date	Total Leakage (Note)	Acceptance Limit (Note)
8/91	0.020%	< 0.075%
2/98	0.065%	< 0.075%

Table 4.1.1d
Byron Station Unit 2 ILRT Test Results
(% weight per day)

ILRT Test Date	Total Leakage (Note)	Acceptance Limit (Note)
8/93	0.067%	< 0.075%
11/99	0.071%	< 0.075%

Note: Leakage rates are expressed in units of containment air weight percent per day at test pressure equal to the calculated peak containment internal pressure related to the Design Basis Accident. Calculated results are expressed at a 95% confidence level plus leakage attributed to non-vented penetrations. The maximum allowable containment leakage rate allowed by Option B during containment leak rate testing was 0.075 weight percent per day ($1.0L_a$).

By letter and Safety Evaluation dated September 8, 2006, the NRC issued License Amendment 140 for Braidwood Station Units 1 and 2 and License Amendment 147 for Byron Station Units 1 and 2. These amendments fully implemented an alternative source term pursuant to the requirements of 10 CFR 50.67, "Accident source term." The License Amendments, in part, revised the maximum allowable containment leakage rate (i.e., L_a at P_a) in TS 5.5.16 for both Braidwood Station and Byron Station from 0.10% of containment air weight per day to 0.20%.

This increase in the maximum allowable containment leakage rate will increase the acceptance criteria for all future ILRT results from < 0.075 % to < 0.15%. This provides additional assurance that the Braidwood Station and Byron Station containment structures will continue to perform their design function following an accident, up to and including the design basis accident.

4.1.2 Type B and C Testing

Type B and C testing at Braidwood Station and Byron Station ensures that containment penetrations such as air locks, flanges, sealing mechanisms and containment isolation valves are essentially leak tight.

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

The Type B and C testing requirements will not be changed as a result of the proposed license amendment.

4.1.3 Containment Inspections

4.1.3.1 Visual Inspections

As part of the Appendix J Program, both Braidwood Station and Byron Station perform visual inspections of accessible interior and exterior surfaces of the containment system for structural problems that may affect either the containment structural leakage integrity or performance of the Type A Test. These examinations are conducted in accordance with the requirements of 10 CFR 50 Appendix J, RG 1.163, and NEI 94-01, prior to any Type A test, to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness, and during the two refueling outages before the next Type A test, based on a ten-year frequency. Additional visual inspections are conducted in accordance with the requirements of American Society of Mechanical Engineers (ASME) Section XI, "Inservice Inspection," Subsection IWE, "Requirements for Class NMC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," (ASME IWE) and Subsection IWL, "Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants," (ASME IWL)

Since the last Type A test, the most recent visual inspections of the accessible interior and exterior surfaces of the Braidwood Station and Byron Station containments were completed during the following years:

Table 4.1.3.1 Most Recent Visual Inspections	
	Date
Braidwood Station Unit 1	2006
Braidwood Station Unit 2	2006
Byron Station Unit 1	2006
Byron Station Unit 2	2004

These visual inspections indicated that that there were no structural problems that could have affected the containment structural leakage integrity. The Appendix J visual inspection frequency, in accordance with RG 1.163 requirements, will not be changed as a result of the proposed change.

4.1.3.2 IWE and IWL Containment Inservice Inspection Program

A comprehensive containment inspection is performed at Braidwood Station and Byron Station in accordance with the requirements of ASME IWE and IWL.

The acceptance criteria used for the examination of IWE and IWL components are established by EGC and comply with Subsections IWE-3000 and IWL-3000 of the ASME code, respectively.

The Braidwood Station and Byron Station Containment Inservice Inspection (CISI) Programs were developed in accordance with the 1992 Edition, 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE and IWL, as modified by NRC final rulemaking to 10 CFR 50.55, which was published in the Federal register on August 8, 1996.

The initial inspections and tests (i.e., first period ASME IWE and first ASME IWL examinations) were completed at both Braidwood Station Units 1 and 2 and Byron Station Units 1 and 2 prior to September 2001. Since 2001, inspections and tests have been completed in accordance with the frequency specified in ASME IWE and IWL. During these inspections, EGC identified various indications that were either repaired, or documented and evaluated as acceptable by the Responsible Individual and Responsible Engineer, with no loss of structural integrity.

As part of the ASME IWL-2521 inspections and tests of post-tensioned tendon systems at Braidwood Station and Byron Station, EGC verified that all pre-stressed forces for the selected tendons met the acceptance criteria for predicted forces. EGC also completed regression analyses after both the post-tensioned tendon visual inspections and physical tests, as described in NRC Information Notice 99-10 Revision 1, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments." The regression analyses indicated acceptable margin for all tendons for the design life of Braidwood Station and Byron Station.

In addition to the required ASME IWL-2521 examinations and tests of post-tensioned tendons at Braidwood Station, EGC conducts an annual inspection at Braidwood Station of grease cans located in areas that are susceptible to moisture intrusion. The scope of the annual examinations includes the tendon grease cans located below grade level and the dome tendons.

The IWL and IWE program examinations have demonstrated that the structural integrity and leak-tightness of the Braidwood Station and Byron Station containment have not been compromised.

There will be no change to the CISI schedule at Braidwood Station and Byron Station as a result of the proposed changes. EGC will conduct CISI inspections of the containments at Braidwood Station Units 1 and 2 and Byron Station Units 1 and 2, as originally scheduled per the CISI Program.

4.1.3.3 Coatings Inspections

EGC conducts periodic inspections of Service Level 1 coatings inside containment during each refueling outage at Braidwood Station and Byron Station, as required by the plant licensing basis and station procedures. These inspections meet the requirements of RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." As localized areas of degraded coatings are identified, those areas are evaluated and scheduled for repair or replacement, as necessary.

Recent inspections of containment coatings at Braidwood Station and Byron Station indicate that although there were some instances of degraded coatings on the liner plate and other surfaces, containment coatings were generally in good condition. In areas where indications were observed, EGC implemented appropriate measures to evaluate and repair the degraded areas. There were no imminent concerns of coating deterioration that would affect the safe operation or safe shutdown of the plants.

The inspection requirements of the containment coatings program at Braidwood Station and Byron Station will not be changed as a result of the proposed changes, including scheduled coating inspections for the upcoming refueling outages.

4.1.3.4 Maintenance Rule Inspections

Maintenance Rule Baseline Inspections required by 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," were completed for Braidwood Station and Byron Station. The inspections included the internal containment structures. Based upon these baseline inspections, EGC concluded that these structures are being adequately maintained and capable of performing their intended functions. This program ensures that internal containment structures at Braidwood Station and Byron Station are evaluated and maintained in condition to perform their intended functions. There will be no changes to the Maintenance Rule Program as a result of the proposed changes.

4.2 Risk Assessments

EGC has conducted risk assessments to determine the impact of a change to the Braidwood Station and Byron Station Type A test schedule from a baseline value of once in 10 years to once in 15 years for the risk measures of Large Early Release Frequency (LERF), Total Population Dose, and Conditional Containment Failure Probability (CCFP). The risk assessments for Braidwood Station and Byron Station are provided in Attachments 4 and 5, respectively.

The risk assessments were developed utilizing the guidance provided in NEI 94-01, RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," the methodology described in EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994, the interim guidance provided by NEI concerning performance of risk impact assessments in support of one-time extensions for containment integrated leakage rate test intervals (References 1 and 2), and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval (Reference 3). The format of the risk assessments is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the December 2005 EPRI final report, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals."

The NRC has previously reviewed and approved the application of the guidance provided in NEI 94-01, RG 1.174, the methodology described in EPRI TR-104285, and the guidance described in References 1, 2, and 3 in the development of risk assessments to justify one-time extensions for containment integrated leakage rate test intervals (i.e., as listed in Section 7.0, "Precedent").

The impact of the analyzed extension to the Braidwood Station and Byron Station Type A test schedule has a minimal impact on public risk, as described below in Sections 4.2.1 through 4.2.4.

4.2.1 LERF

The increases in LERF resulting from a Type A test schedule extension from once in 10 years to once in 15 years is described in the Table 4.2.1 below.

TABLE 4.2.1
Increase in LERF due to Extension of Type A Test Interval

	Current LERF (once in 10 years)	**Revised LERF (once in 15 years)**	**Δ LERF**
Braidwood 1	4.54E-07/yr	6.81E-07/yr	2.27E-07/yr
Braidwood 2	4.46E-07/yr	6.70E-07/yr	2.24E-07/yr
Byron 1	4.88E-07/yr	7.35E-07/yr	2.47E-07/yr
Byron 2	4.83E-07/yr	7.27E-07/yr	2.44E-07/yr

RG 1.174 defines small changes in LERF as increases less than 1.0E-06/yr, but greater than 1.0E-07/yr. The RG also states that a small change in LERF is acceptable provided that the total LERF is less than 1.0E-05/yr. The Δ LERF values associated with the proposed Type A test schedule extension in Table 4.2.1 are within the criteria for a

"small change" established by RG 1.174, and the total LERF is less than 1.0E-05. Therefore, the increase in LERF is acceptable.

4.2.2 Total Population Dose

The increase in total integrated plant risk (i.e., total population dose) resulting from a Type A test schedule extension from once in 10 years to once in 15 years is described in the Table 4.2.2 below. These values impact only those accident sequences influenced by Type A testing.

TABLE 4.2.2
Increase in total Population Dose due to Extension of Type A Test Interval

	Current Total Dose (person-rem/yr) (once in 10 years)	**Revised Total Dose (person-rem/yr) (once in 15 years)**	**Δ Total Dose (person-rem/yr)**
Braidwood 1	16.13	16.19	0.06
Braidwood 2	17.66	17.72	0.06
Byron 1	3.64	3.65	0.01
Byron 2	4.09	4.11	0.02

4.2.3 CCFP

The increase in CCFP resulting from a Type A test schedule extension from once in 10 years to once in 15 years is described in Table 4.2.3 below. These increases are insignificant relative to the current CCFP for an ILRT interval of once in ten years.

TABLE 4.2.3
Increase in CCFP due to Extension of Type A Test Interval

	Current CCFP (once in 10 years)	**Revised CCFP (once in 15 years)**	**Δ CCFP**
Braidwood 1	9.96%	10.37%	0.41%
Braidwood 2	11.51%	11.93%	0.42%
Byron 1	9.00%	9.42%	0.42%
Byron 2	10.65%	11.07%	0.42%

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

The risk assessments provided in Attachments 4 and 5 include an evaluation of the likelihood and risk implications of undetected corrosion-induced leakage of the steel liners during the extended test interval (i.e., 15 years). This evaluation utilized the same methodology used in the Calvert Cliffs liner corrosion analysis (Reference 3). The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. Both Braidwood Station and Byron Station have a similar type of containment structure.

Key assumptions in the evaluation of the likelihood and risk implications of undetected corrosion-induced leakage of the steel liners were as follows:

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures.
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to the Braidwood Station and Byron Station containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2 were initiated from the non-visible (backside) portion of the containment liner.
- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is calculated using a 5.5-year data period to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data were not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date and there is no evidence that additional corrosion issues were identified.
- Consistent with the Calvert Cliffs analysis, the corrosion-induced steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages. Sensitivity studies are included in the risk assessment that address doubling this rate every ten years and every two years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated (based on an assessment of the containment fragility curve versus the ILRT test pressure) as 1.1% for the cylinder and dome region and 0.11% (10% of the cylinder failure probability) for the basemat. Similarly, for Braidwood Station and Byron Station, the containment failure probabilities are conservatively assumed to be 1% for the cylinder and dome, and 0.1% for the basemat.
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a 10% likelihood of a non-detectable flaw is used. Again, this is considered conservative since to date, all liner corrosion events have been detected through visual inspection. Sensitivity studies are included in the risk assessments that evaluate total detection failure likelihood of 5% and 15%.
- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

The impact from including the corrosion effects in the base case analyses is very minimal and was included in the results shown in Table 4.2.1 through 4.2.3. The upper bound potential impact of corrosion effects resulting from a Type A test schedule extension measured from the original three in ten year testing interval is summarized in Table 4.2.4 below. The results indicate that even with very conservative assumptions, the conclusions from the base analyses would not change.

Table 4.2.4 **Upper Bound Δ LERF Due to Corrosion Effects**	
Braidwood 1	1.75E-07/yr
Braidwood 2	1.73E-07/yr
Byron 1	1.88E-07/yr
Byron 2	1.86E-07/yr

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC, (EGC), requests an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station Units 1 and 2, respectively. Specifically, the proposed changes will revise TS 5.5.16, "Containment Leakage Rate Testing Program," to reflect a one-time, five-year extension of the current containment Type A test date requirement.

According to 10 CFR 50.92, "Issuance of amendment," Section (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or
3. Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes will revise Braidwood Station and Byron Station TS 5.5.16, "Containment Leakage Rate Testing Program" to reflect a one-time, five-year extension of the containment Type A test date to enable the implementation of a 15-year test interval.

The containment is designed to contain radioactive material that may be released from the reactor core following a design basis Loss of Coolant Accident (LOCA). The test interval associated with Type A testing is not a precursor of any accident previously evaluated. Type A testing does provide assurance that the containment will not exceed allowable leakage rate criteria specified in the TS and will continue to perform its design function following an accident. A risk assessment of the proposed changes has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes for a one-time, five-year extension of the Type A tests for Braidwood Station and Byron Station will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed changes do not introduce any new equipment, modes of system operation or failure mechanisms.

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

3. The proposed changes do not involve a significant reduction in a margin of safety.

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

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. The cylinder wall is pre-stressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is pre-stressed utilizing a three way post-tensioning system.

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

The integrity of the containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak tight characteristics of the containment at the design basis accident pressure.

The existing 10-year Type A test interval is based on past performance. Previous Type A leakage tests conducted at Braidwood Station Units 1 and 2, and Byron Station Units 1 and 2 indicate that leakage from containment has been less than the 10 CFR 50 Appendix J leakage limit.

The proposed changes for a one-time extension of the Type A tests do not affect the method for Type A, B or C testing or the test acceptance criteria. Type B and C testing will continue to be performed at the frequency required by the Braidwood Station and

Byron Station Technical Specifications. The containment inspections that are performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI and 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," provide a high degree of assurance that the containment will not degrade in a manner that is only detectable by Type A testing.

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

Therefore, based on previous Type A test results for the Braidwood Station and Byron Station containments, the current containment surveillance programs at each station, and the results of the EGC risk assessments, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the above discussion, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Regulatory Requirements & Guidance

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

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

The proposed changes will revise Braidwood Station and Byron Station TS Section 5.5.16 to reflect a one-time extension from the program requirements for the Type A test. The one-time extension deviates from the guidelines contained in RG 1.163 and NEI 94-01. Thus, the proposed changes are consistent with the requirements of 10 CFR 36(c)(5) and 10 CFR 50, Appendix J, Section V. B.

Additionally, in accordance with 10 CFR 50, Appendix J, Section V. Option B, the proposed changes to Braidwood Station and Byron Station TS do not require a supporting request for an exemption to Option B of Appendix J, in accordance with 10 CFR 50.12, "Specific exemptions."

6.0 ENVIRONMENTAL CONSIDERATION

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

1. The amendment involves no significant hazards consideration.

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

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

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

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

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

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

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENT

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

- Indian Point Nuclear Generating Unit No. 3 (April 17, 2001)
- Crystal River Unit 3 (August 30, 2001)
- Peach Bottom Atomic Power Station, Unit 3 (October 4, 2001)
- Oconee Nuclear Station, Unit 3 (February 28, 2002)
- Susquehanna Steam Electric Station, Units 1 and 2 (March 8, 2002)
- Seabrook Station, Unit No. 1 (April 11, 2002)
- Calvert Cliffs Nuclear Power Plant, Unit No. 1 (May 1, 2002)
- Indian Point Nuclear Generating Unit No. 2 (August 5, 2002)
- Comanche Peak Steam Electric Station, Units 1 and 2 (August 15, 2002)
- North Anna Power Station, Unit 1 (December 31, 2002)
- Beaver Valley Power Station, Unit Nos. 1 and 2 (March 5, 2003)
- River Bend Station, Unit 1 (March 5, 2003)
- McGuire Nuclear Station, Units 1 and 2 (March 12, 2003)
- Catawba Nuclear Station, Units 1 and 2 (March 12, 2003)
- Joseph M. Farley Nuclear Plant, Units 1 and 2 (March 21, 2003)
- Duane Arnold Energy Center (March 21, 2003)
- Monticello Nuclear Generating Plant (March 31, 2003)
- Perry Nuclear Power Plant, Unit 1 (April 8, 2003)
- Hope Creek Generating Station (April 16, 2003)
- Sequoyah Nuclear Plant, Units 1 and 2 (May 29, 2003)
- Three Mile Island Nuclear Station, Unit 1 (August 14, 2003)
- Fort Calhoun Station, Unit No. 1 (August 15, 2003)
- LaSalle County Station, Units 1 and 2 (November 19, 2003)
- Clinton Power Station (January 8, 2004)
- Vogtle Electric Generating Plant, Units 1 and 2 (January 12, 2004)
- Grand Gulf Nuclear Station, Unit 1 (January 28, 2004)
- H. B. Robinson Steam Electric Plant, Unit No. 2 (February 11, 2004)
- Quad Cities Nuclear Power Station, Units 1 and 2 (March 8, 2004)
- Kewaunee Nuclear Power Plant (April 6, 2004)
- James A. Fitzpatrick Nuclear Power Plant (September 28, 2004)
- Dresden Nuclear Power Station, Units 2 and 3 (October 13, 2004)
- Edwin I. Hatch Nuclear Plant, Unit 2 (February 1, 2005)
- Browns Ferry Nuclear Plant, Units 2 and 3 (March 9, 2005)

- Pilgrim Nuclear Power Station (March 30, 2005)
- Millstone Power Station, Unit No. 2 (April 6, 2005)
- Columbia Generating Station (April 12, 2005)
- Vermont Yankee Nuclear Power Station (August 31, 2005)
- R. E. Ginna Nuclear Power Plant (December 8, 2005)
- St. Lucie Plant, Unit No. 2 (December 23, 2005)
- Shearon Harris Nuclear Power Plant, Unit 1 (March 30, 2006)
- Watts Bar Nuclear Plant, Unit 1 (August 22, 2006)

8.0 REFERENCES

1. Letter from A. Pietrangelo (NEI) to NEI Administrative Points-of-Contact, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals," dated November 13, 2001.
2. Letter from A. Pietrangelo (NEI) to NEI Administrative Points-of-Contact, "One-Time Extension of Containment Integrated Leak Rate Test Interval – Additional Information," dated November 30, 2001.
3. Letter from C.H. Cruse (Calvert Cliffs Nuclear Power Plant) to USNRC, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," dated March 27, 2002.

ATTACHMENT 2A

Braidwood Station
Units 1 and 2

NRC Docket Nos. 50-456 and 50-457

Facility Operating License Nos. NPF-72 and NPF-77

Mark-up of Technical Specification Page

TS 5.5-16

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

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

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

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

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 criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

INSERT 1

INSERT 1

as modified by the following exceptions:

1. NEI 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after the October 5, 1998 Type A test shall be performed no later than October 5, 2013.
2. NEI 94-01 – 1995, Section 9.2.3: The first Unit 2 Type A test performed after the May 4, 1999 Type A test shall be performed no later than May 4, 2014.

ATTACHMENT 2B

Byron Station
Units 1 and 2

NRC Docket Nos. 50-454 and 50-455

Facility Operating License Nos. NPF-37 and NPF-66

Mark-up of Technical Specification Page

TS 5.5-16

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

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

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

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

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 criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

INSERT 2

INSERT 2

as modified by the following exceptions:

1. NEI 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after the February 19, 1998 Type A test shall be performed no later than February 19, 2013.
2. NEI 94-01 – 1995, Section 9.2.3: The first Unit 2 Type A test performed after the November 2, 1999 Type A test shall be performed no later than November 2, 2014.

ATTACHMENT 3A

Braidwood Station
Units 1 and 2

NRC Docket Nos. 50-456 and 50-457

Facility Operating License Nos. NPF-72 and NPF-77

Retyped Technical Specification Page

TS 5.5-16

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

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

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0 as modified by the following exceptions:

1. NEI 94-01 - 1995, Section 9.2.3: The first Unit 1 Type A test performed after the October 5, 1998 Type A test shall be performed no later than October 5, 2013.

2. NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after the May 4, 1999 Type A test shall be performed no later than May 4, 2014.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

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

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 criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

ATTACHMENT 3B

Byron Station
Units 1 and 2

NRC Docket Nos. 50-454 and 50-455

Facility Operating License Nos. NPF-37 and NPF-66

Retyped Technical Specification Page

TS 5.5-16

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

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

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0 as modified by the following exceptions:

1. NEI 94-01 - 1995, Section 9.2.3: The first Unit 1 Type A test performed after the February 19, 1998 Type A test shall be performed no later than February 19, 2013.

2. NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after the November 2, 1999 Type A test shall be performed no later than November 2, 2014.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

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 criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

ATTACHMENT 4

Risk Assessment for Braidwood Unit 1 and Unit 2 to Support ILRT (Type A) Interval Extension Request

RM DOCUMENTATION NO.	BB PRA-017.54A	REV: 3	PAGE NO. 1

STATION: BRAIDWOOD

UNIT(S) AFFECTED: 1 and 2

TITLE: Risk Assessment for Braidwood Unit 1 and Unit 2 To Support ILRT (Type A) Interval Extension Request

SUMMARY (Include UREs incorporated): The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Braidwood Unit 1 and Braidwood Unit 2 containment Type A integrated leak rate test (ILRT) interval from 10 years to 15 years.

Revision 3 removed incorporation of specific ILRT test results and changed verb tense in background section.

Internal RM Documentation

Electronic Calculation Data Files:
(Program Name, Version, File Name extension/size/date/hour/min)

Refer to Appendix A of the document.

Main Body

Prepared by: Donald E. Vanover / Donald E. Vanover / 3/2/07
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Appendix A

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This RM documentation supersedes: Rev. 2 **in its entirety.**

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Print / Sign / Date

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Do any ASSUMPTIONS / ENGINEERING JUDGEMENTS require later verification? **[ ] Yes [X] No**
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TABLE OF CONTENTS

Section **Page**

1.0 PURPOSE OF ANALYSIS 3
1.1 Purpose 3
1.2 Background 3
1.3 Criteria 5
2.0 METHODOLOGY 6
3.0 GROUND RULES 8
4.0 INPUTS 10
4.1 General Resources Available 10
4.2 Plant-Specific Inputs 18
4.3 Impact of Extension on Detection of Component Failures That Lead to Leakage (Small and Large) 24
4.4 Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage 27
5.0 RESULTS 34
5.1 Step 1 – Quantify the Base-Line Risk in Terms of Frequency per Reactor Year 36
5.2 Step 2 – Develop Plant-Specific Person-Rem Dose (Population Dose) per Reactor Year 41
5.3 Step 3 – Evaluate Risk Impact of Extending Type A Test Interval From 10-to-15 Years 45
5.4 Step 4 – Determine the Change in Risk in Terms of Large Early Release Frequency 51
5.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability 52
5.6 Summary of Results 53
6.0 SENSITIVITIES 56
6.1 Sensitivity to Corrosion Impact Assumptions 56
6.2 EPRI Expert Elicitation Sensitivity 58
6.3 Potential Impact from External Events Contribution 62
7.0 CONCLUSIONS 64
8.0 REFERENCES 67
Appendix A CDF and LERF Subcategory Calculations 70

1.0 PURPOSE OF ANALYSIS

1.1 Purpose

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Braidwood Station Units 1 and 2 containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3, 21], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [19]. The format of this document is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the December 2005 EPRI final report [22].

1.2 Background

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is 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 normal containment leakage of $1.0L_a$ (allowable leakage).

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [5],

"Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285.

The NRC report, Performance Based Leak Test Program, NUREG-1493 [5], analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis contained herein has been performed to confirm these conclusions based on Braidwood specific models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In November and December 2001, NEI issued enhanced guidance (hereafter referred to as the NEI Interim Guidance) that builds on the TR-104285 methodology and intended to provide for more consistent submittals. [3,21] The NEI Interim Guidance was developed for NEI by EPRI using personnel who also developed the TR-104285 methodology. This ILRT interval extension risk assessment for Braidwood employs the NEI Interim Guidance methodology.

It should be noted that, in addition to ILRT tests, containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and

their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment 3 times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency. Type C tests are also not affected by the Type A test frequency change.

1.3 Criteria

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

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. This is based on the precedent set by previous submittals [6, 20, 23]. (No criteria have been established for this parameter change.)

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years [22]. The approach is consistent with that presented in NEI Interim Guidance [3, 21], EPRI TR-104285 [2], NUREG-1493 [5] and the Calvert Cliffs liner corrosion analysis [19]. The analysis uses results from a Level 2 analysis of core damage scenarios from the current Braidwood PRA model and subsequent containment response resulting in LERF and non-LERF endstates. This risk assessment is applicable to Braidwood Units 1 and 2.

The six general steps of this assessment are as follows:

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

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore,

- Consistent with the other industry containment leak risk assessments, the Braidwood assessment uses population dose as one of the risk measures. The other risk measures used in the Braidwood assessment are LERF and the conditional containment failure probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.
- This evaluation for Braidwood uses ground rules and methods to calculate changes in risk metrics that are similar to those used in the EPRI approach.

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The Braidwood Level 1 and LERF internal events PRA models provide representative results.
- It is appropriate to use the Braidwood internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations.
- Dose results for the containment failures modeled in the PRA can be characterized by information provided in NUREG/CR-4551 [9]. They are estimated by scaling the NUREG/CR-4551 results by population differences for Braidwood compared to the NUREG/CR-4551 reference plant.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is $1L_a$. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is $10L_a$, based on the previously approved methodology performed for Indian Point Unit 3 [6, 7].
- The representative containment leakage for Class 3b sequences is $35L_a$, based on the previously approved methodology [6, 7].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [6, 7]. The Class 3b category increase is used as a surrogate for LERF in this application even though the releases associated with a $35L_a$ release would not necessarily be consistent with a "Large" release for Braidwood. (See, for example, the calculated population dose results for EPRI Class 3b in Table 5.2-1 of 7.53E3 person-rem compared to the 3.5E6 person-rem associated with EPRI Class 8 for containment bypass scenarios.)

- For simplicity, and since Braidwood only tracks LERF versus non-LERF release categories, all non-LERF endstates are assumed to be in EPRI Class 1. This may tend to under-predict the calculated plant-specific total dose and CCFP values, but will not change the calculated changes in population dose, LERF, and CCFP that are the determined in this risk impact assessment.
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The use of estimated 2010 population data is adequate for this analysis. Scaling the year 2010 population data to the date of the next ILRT test if extended beyond the current due date would not significantly impact the quantitative results, nor would it change the conclusions.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8].

4.0 INPUTS

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

4.1 General Resources Available

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

1. NUREG/CR-3539 [10]
2. NUREG/CR-4220 [11]
3. NUREG-1273 [12]
4. NUREG/CR-4330 [13]
5. EPRI TR-105189 [8]
6. NUREG-1493 [5]
7. EPRI TR-104285 [2]
8. NUREG-1150 [14] and NUREG/CR-4551 [9]
9. NEI Interim Guidance [3, 21]
10. Calvert Cliffs liner corrosion analysis [19]
11. EPRI 1009325 [22]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different

containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the bases for the consequence analysis of the ILRT interval extension for Braidwood. The ninth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the last study complements the previous EPRI report [2], integrates the NEI interim guidance, and provides the results of an expert elicitation process to determine the relationship between pre-existing containment leakage probability and magnitude.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [15] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories (PNL) for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage. The study calculated unavailabilities for Technical Specification leakages and "large" leakages. NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 PWR events in 740 reactor years and conservatively assuming a one-year duration for each event.

NUREG-1273 [12]

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

NUREG/CR-4330 [13]

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

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

EPRI TR-105189 [8]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk.

The result of the study concluded that a small but measurable safety benefit (shutdown CDF reduced by 1E-8/yr to 1E-7/yr) is realized from extending the test interval from 3 per 10 years to 1 per 10 years.

NUREG-1493 [5]

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

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2 – 1.0%) population risk.
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

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

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage sequences into eight categories of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures

7. Containment failure due to core damage accident phenomena
8. Containment bypass

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

> *"These study results show that the proposed CLRT* [containment leak rate tests] *frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year..."*

Release Category Definitions

Table 4.1-1 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI/NEI methodology [2]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

Table 4.1-1

EPRI/NEI CONTAINMENT FAILURE CLASSIFICATIONS [2]

Class	Description
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

NUREG-1150 [14] and NUREG/CR 4551 [9]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding the Surry Power Station. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the Braidwood LERF end-

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

NEI Interim Guidance [3, 21]

NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions of Containment Integrated Leakage Rate Test Surveillance Intervals" [3] has been developed to provide utilities with revised guidance regarding licensing submittals. Additional information from NEI on the "Interim Guidance" was supplied in Reference [21].

A nine step process is defined which includes changes in the following areas of the previous EPRI guidance:

- Impact of extending surveillance intervals on dose
- Method used to calculate the frequencies of leakages detectable only by ILRTs
- Provisions for using NUREG-1150 dose calculations to support the population dose determination.

The guidance provided in this document builds on the EPRI risk impact assessment methodology [2] and the NRC performance-based containment leakage test program [5], and considers approaches utilized in various submittals, including Indian Point 3 [6,7] (and associated NRC SER) and Crystal River [20].

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

Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [19].

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

EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [22]

This report presents a risk impact assessment for extending integrated leak rate test (ILRT) surveillance intervals to 15 years and is consistent in nature with the NEI interim guidance. This risk impact assessment complements the previous EPRI report, TR-104285, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals.* The earlier report considered changes to local leak rate testing intervals as well as changes to ILRT testing intervals. The original risk impact assessment considers the change in risk based on population dose, whereas the revision considers dose as well as large early release frequency (LERF) and conditional containment failure probability (CCFP). This report deals with changes to ILRT testing intervals and is intended to provide bases for supporting changes to industry (NEI) and regulatory (NRC) guidance on ILRT surveillance intervals.

The risk impact assessment using the Jeffery's Non-Informative Prior statistical method is further supplemented with a sensitivity case using expert elicitation performed to address conservatisms. The expert elicitation is used to determine the relationship between pre-existing containment leakage probability and magnitude. The results of the expert elicitation process from this report are used as a separate sensitivity investigation for the Braidwood analysis presented here in Section 6.2.

4.2 Plant-Specific Inputs

The Braidwood specific information used to perform this ILRT interval extension risk assessment includes the following:

- Level 1 Model results [16]
- LERF Model results [16]
- Population within a 50-mile radius [18]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [(1)]

Braidwood Internal Events Level 1 PRA

The Braidwood Level 1 PRA used as input to this analysis is characteristic of the as-built, as-operated plant. The current Level 1 PRA model is a linked fault tree model characteristic of the as-built plant. The total internal events core damage frequency (CDF) used in this analysis is 5.46E-05/yr for Unit 1 and 5.38E-05/yr for Unit 2 [16].

Braidwood Internal Events LERF Model

Table 4.2-1 summarizes the pertinent Braidwood results in terms of EPRI/NEI accident class and NUREG/CR-4551 accident progression bin. The total internal events large early release frequency (LERF) used in this analysis is 4.99E-06/yr for Unit 1 and 5.75E-06/yr for Unit 2. For simplicity, and since Braidwood only tracks LERF versus non-LERF release categories, as detailed in Section 5.1 of this report all non-LERF endstates are assumed to be in EPRI Class 1. This may tend to under-predict the calculated plant-specific total dose and CCFP values, but will not change the calculated changes in population dose, LERF, and CCFP that are the determined in this risk impact assessment.

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(1) The two most recent Type A tests at Braidwood Unit 1 and Unit 2 have been successful, so the current Type A test interval requirement is 10 years [24].

Table 4.2-1 Braidwood LERF Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins and EPRI / NEI Accident Classes

Braidwood LERF Category[1]	Definition	NUREG/ CR-4551 APB	EPRI / NEI Class	Unit 1 Frequency /yr	Unit 2 Frequency/ yr
A	Straight pass through CDF sequence to LERF	5	8	3.43E-06	3.35E-06
B_1	High pressure sequences with no AFW available.	2	2	4.91E-08	4.23E-08
B_2	High pressure sequences with no AFW available, where the possibility exists for an ISGTR.	5	8	8.42E-08	7.91E-08
B_3	High pressure sequences with no AFW available and questions whether the containment fails at the time of vessel breach.	2	7	1.19E-07	2.29E-07
C_1	High pressure sequences with no AFW available.	2	2	3.23E-10	3.23E-10
C_2	High pressure sequences with no AFW available, where the possibility exists for an ISGTR.	5	8	2.91E-08	2.91E-08
C_3	High pressure sequences with no AFW available and the time between core uncovery and vessel breach is greater than the required evacuation time.	2	7	0	0
D	Sequences that do not lead to containment failure or result in containment failure many hours after core uncovery. Containment isolation is asked.	2	2	5.83E-07	6.47E-07
E	SGTR sequences where isolation of the ruptured SG is possible but has not been questioned in the Level 1 event tree, and asks to see isolation of the ruptured SG is successful.	5	8	5.42E-09	2.11E-10
F_1	High pressure sequences with AFW available.	2	2	4.12E-09	6.64E-09
F_2	High pressure sequences with AFW available, where no possibility exists for an ISGTR.	2	7	6.80E-07	1.36E-06
G	Sequences where the RCS pressure is low at the time of vessel breach and AFW available, where no possibility exists for an ISGTR.	3	2	6.06E-09	6.06E-09

[1] Braidwood LERF categories are indicated by the letters. Where a given Braidwood LERF category comprises more than one NUREG APB or EPRI category, subscripts have been assigned in this study to denote the portions of the LERF categories that correspond to the noted NUREG APBs or EPRI categories. The determination of these frequencies for Braidwood Unit 1 and Unit 2 for use in this analysis is described in Appendix A.

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for Braidwood. Each of the release categories from Table 4.2-1 was associated with an applicable Collapsed Accident Progression Bin (APB) from NUREG/CR-4551 (see below). The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 7 bins that are relevant to the analysis. The definitions of the 7 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 4.2-2 for references purposes. Table 4.2-3 then summarizes the calculated population dose for Surry associated with each APB from NUREG/CR-4551.

Table 4.2-2

Summary Accident Progression Bin (APB) Descriptions [9]

Summary APB Number	Summary Accident Progression Bin (APB Description
1	CD, VB, Early CF, Alpha Mode Core damage occurs followed by a very energetic molten fuel-coolant interaction in the vessel; the vessel fails and generates a missile that fails the containment as well. Includes accidents that have an Alpha mode failure of the vessel and the containment except those that follow Event V or an SGTR. It includes Alpha mode failures that follow isolation failures because the Alpha mode containment failure is of rupture size.
2	CD, VB, Early CF, RCS Pressure > 200psia Core Damage occurs followed by vessel breach. Implies Early CF with the RCS above 200 psia when the vessel fails. Early CF means at or before VB, so it includes isolation failures and seismic containment failures at the start of the accident as well as containment failure at VB. It does not include bins in which containment failure at VB follows Event V or an SGTR, or Alpha mode failures.
3	CD, VB, Early CF, RCS Pressure < 200 psia Core damage occurs followed by vessel breach. Implies Early CF with the RCS below psia when the containment fails. It does not include bins in which the containment fails at VB, an SGTR, or Alpha mode failures.
4	CD, VB, Late CF Core Damage occurs followed by vessel breach. Includes accidents in which the containment was not failed or bypassed before the onset of core-concrete interaction (CCI) and in which the vessel failed. The failure mechanisms are hydrogen combustion during CCI, Basemat Melt-Through (BMT) in several days, or eventual overpressure due to the failure to provide containment heat removal in the days following the accident.
5	CD, Bypass Core Damage occurs followed by vessel breach. Includes Event V and SGTRs no matter what happens to the containment after the start of the accident. It also includes SGTRs that do not result in VB.
6	CD, VB, No CF Core Damage occurs followed by vessel breach. Includes accidents not evaluated in one of the previous bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.
7	CD, No VB Core Damage occurs but is arrested in time to prevent vessel breach. Includes accident progressions that avoid vessel failures except those that bypass the containment. Most of the bins placed in this bin have no containment failure as well as no VB. It also includes bins in which the containment is not isolated at the start of the accident and the core is brought to a safe stable state before the vessel fails.

Table 4.2-3
Calculation of Surry Population Dose Risk at 50 Miles [9]

Collapsed Bin #	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (person-rem/yr, mean) (2)	NUREG/CR-4551 Collapsed Bin Frequencies (per year) (3)	NUREG/CR-4551 Population Dose at 50 miles (person-rem) (4)
1	0.029	0.158	1.23E-07	1.28E+06
2	0.019	0.106	1.64E-07	6.46E+05
3	0.002	0.013	2.01E-08	6.46E+05 (5)
4	0.216	1.199	2.42E-06	4.95E+05
5	0.732	4.060	5.00E-06	8.12E+05
6	0.001	0.006	1.42E-05	4.23E+02
7	0.002	0.011	1.91E-05	5.76E+02
Totals	1.000	5.55	4.1E-05	

(1) Mean Fractional Contribution to Risk calculated from the average of two samples delineated in Table 5.1-3 of NUREG/CR-4551.

(2) The total population dose risk at 50 miles from internal events in person-rem is provided as the average of two samples in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.

(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-3. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.

(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.

(5) Assumed population dose at 50 miles for Collapsed Bin #3 equal to that of Collapsed Bin #2. Collapsed Bin Frequency #3 was then back calculated using that value. This does not influence the results of this evaluation since Bin #3 does not appear as part of the results for Braidwood.

Population Estimate Methodology

The person-rem results in Table 4.2-3 can be used as an approximation of the dose for Braidwood if it is corrected for the population surrounding Braidwood. The total population within a 50-mile radius of Braidwood is projected to be 5.304E+06 by the year 2010 [18]. This value is slightly more than the projected value of 4.973E+06 from the Braidwood UFSAR since it factors in more recent actual census data from 1990 and 2000 for the projected growth estimates compared to the earlier population data utilized in the UFSAR. The use of the 2010 estimate is judged to be sufficient to perform this assessment. Scaling the year 2010 population data to the date of the next ILRT test if extended beyond the current due date would not significantly impact the quantitative results, nor would it change the conclusions.

This population value is compared to the population value that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for Braidwood.

Total Braidwood $\text{Population}_{50\text{miles}}$ = 5.304E+06
Surry Population from NUREG/CR-4551 = 1.23E+06
Population Dose Factor = 5.304E +06 / 1.23E+06 = 4.31

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. This does not take into account differences in meteorology data, detailed environmental factors or detailed differences in containment designs, but does provide a first-order approximation for Braidwood of the population doses associated with each of the release categories from NUREG/CR-4551. This is considered adequate since the conclusions from this analysis will not be substantially affected by the actual dose values that are used.

Table 4.2-4 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for Braidwood.

Table 4.2-4
Calculation of Braidwood Population Dose Risk at 50 Miles

Accident Progression Bin (APB)	NUREG/CR-4551 Population Dose at 50 miles (person-rem)	Bin Multiplier used to obtain Braidwood Population Dose	Braidwood Adjusted Population Dose at 50 miles (person-rem)
1	1.28E+06	4.31	5.52E+06
2	6.46E+05	4.31	2.79E+06
3	6.46E+05	4.31	2.79E+06
4	4.95E+05	4.31	2.13E+06
5	8.12E+05	4.31	3.50E+06
6	4.23E+02	4.31	1.82E+03
7	5.76E+02	4.31	2.48E+03

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the Braidwood Level 2 model are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in that document, it was necessary to match the Braidwood Level 2 release categories to the collapsed APBs. The assignments are shown in Table 4.2-1, along with the corresponding EPRI/NEI classes.

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

The ILRT can detect a number of component failures such as liner breach, failure of certain bellows arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 4.1-1 is divided into two sub-classes representing small and large leakage failures. These subclasses are defined as Class 3a and Class 3b, respectively.

The probability of the EPRI Class 3a failures may be determined, consistent with the NEI Guidance [3], as the mean failure estimated from the available data (i.e., 5 "small" failures in 182 tests leads to a 5/182=0.027 mean value). For Class 3b, using the original NEI Guidance [3], a non-informative prior distribution would be assumed for no "large" failures in 182 tests (i.e., 0.5/(182+1) = 0.0027).

In a follow on letter [21] to their ILRT guidance document [3], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC regulatory guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> "The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage."

The application of this additional guidance to the analysis for Braidwood, as detailed in Section 5, involves the following:

1. The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the Braidwood Level 2 PRA analysis.

2. Since the Braidwood Level 2 model only includes a representation of LERF vs non-LERF endstates, for simplification all of the non-LERF endstates have been assigned to EPRI Class 1, and as such are included in the CDF multiplier that is subject to the potential impacts from the ILRT extension assessment. In fact, however, a review of Class 1 accident sequences shows that several of these cases could involve successful operation of containment sprays in which case the potential for pre-existing leaks resulting in LERF magnitude releases is greatly reduced. For this assessment, for calculation of the Class 3a and 3b frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. A simplified separate effects containment spray model with a screening Human Error Probability of 0.1 for initiation of the containment sprays (as would be directed by SAG-5, Reduce Fission Product Releases, Rev. 0 [25]) was appended to the existing Level 1 sequence cutsets. This exercise revealed that 50% or more of the EPRI Class 1 CDF could result in sprays being available (even with accounting for dependent operator action failures and all hardware dependencies) thereby reducing the potential for LERF. This potential benefit was conservatively not credited in this analysis.

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

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC [7]) because it does not factor in the possibility that the

failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

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

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [19]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. Braidwood has a similar type of containment.

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

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

<u>Assumptions</u>

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 4.4-1, Step 1.)
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to this Braidwood containment analysis. These events, one at North Anna Unit 2 and one at

Brunswick Unit 2 were initiated from the non-visible (backside) portion of the containment liner.

- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis), and there is no evidence that additional corrosion issues were identified. (See Table 4.4-1, Step 1.)
- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages. (See Table 4.4-1, Steps 2 and 3.) Sensitivity studies are included that address doubling this rate every ten years and every two years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure. For Braidwood, the containment failure probabilities are much less than these values at the target pressures of 42.8 psig for Unit 1 and 38.4 psig for Unit 2. Conservative probabilities of 1% for the cylinder and dome and 0.1% for the basemat are used in this analysis, and sensitivity studies are included that increase and decrease the probabilities by an order of magnitude. (See Table 4.4-1, Step 4.)
- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the containment cylinder and dome region. (See Table 4.4-1, Step 4.)

- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection. (See Table 4.4-1, Step 5.) Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.
- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Table 4.4-1

Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome		Containment Basemat	
1	**Historical Steel Liner Flaw Likelihood**	Events: 2		Events: 0 (assume half a failure)	
	Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).	2/(70 * 5.5) = **5.2E-3**		0.5/(70 * 5.5) = **1.3E-3**	
2	Age Adjusted Steel Liner Flaw Likelihood During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).	Year	Failure Rate	Year	Failure Rate
		1	2.05E-03	1	5.13E-04
		avg 5-10	5.19E-03	avg 5-10	1.30E-03
		15	1.43E-02	15	3.51E-03
		15 year average = 6.14E-03		**15 year average = 1.54E-03**	

Table 4.4-1

Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome	Containment Basemat
3	Flaw Likelihood at 3, 10, and 15 years Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference [19]).	**0.71% (1 to 3 years)** **4.06% (1 to 10 years)** **9.40% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the desired presentation of the results.)	**0.18% (1 to 3 years)** **1.03% (1 to 10 years)** **2.39% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with desired presentation of the results.)
4	Likelihood of Breach in Containment Given Steel Liner Flaw The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.

Table 4.4-1

Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome	Containment Basemat
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0094% (at 15 years)** 9.4% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 100%

Analysis

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

Total Likelihood Of Non-Detected Containment Leakage Due To Corrosion:

- At 3 years: 0.00071% + 0.00018% = 0.00089%
- At 10 years: 0.0041% + 0.0010% = 0.0051%
- At 15 years: 0.0094% + 0.0024% = 0.0118%

Braidwood Past ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 under option B criteria is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3.

Based on completion of two successful ILRTs at each of the Braidwood units, the current ILRT interval is once per ten years. The next Type A test for Braidwood Unit 1 is currently due to be completed by 10/2008, and by 05/2009 for Unit 2.

Note that the probability of a pre-existing leakage due to extending the ILRT interval is based on the industry wide historical results as discussed in the NEI Guidance document, and the only portion of Braidwood specific information utilized is the fact that the current ILRT interval is once per ten years.

NEI Interim Guidance

This analysis uses the approach outlined in the NEI Interim Guidance. [3, 21] The nine steps of the methodology are:

1. Quantify the baseline (nominal three year ILRT interval) frequency per reactor year for the EPRI accident categories of interest. Note that EPRI categories 4, 5, and 6 are not affected by changes in ILRT test frequency.

2. Determine the containment leakage rates for EPRI categories 1 and 3 where category 3 is subdivided into categories 3a and 3b for "small" and "large" isolation failures, respectively.

3. Develop the baseline population dose (person-rem) for the applicable EPRI categories.

4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in Step (3) by the associated frequency calculated in Step (1).

5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.

6. Determine the population dose rate for the new surveillance intervals of interest.

7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.

8. Evaluate the risk impact in terms of LERF.

9. Evaluate the change in conditional containment failure probability.

The first seven steps of the methodology calculate the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The ninth and final step of the interim methodology calculates the change in containment failure probability, referred to the conditional containment failure probability, CCFP. The NRC has previously accepted similar calculations [7] for the change in CCFP as the basis for showing that the proposed change is consistent with the defense in depth philosophy. As such this last step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174.

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "small" leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).

5.0 RESULTS

The application of the approach based on NEI Interim Guidance [3, 21], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6, 7, 20, 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5-1 lists these accident classes.

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

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5-1

ACCIDENT CLASSES

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

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

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5-1.

Step 2 - Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

Step 3 - Evaluate risk impact of extending Type A test interval from 3 to 15 and 10 to 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

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

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

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

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

The frequencies for the severe accident classes defined in Table 5.1-1 were developed for Braidwood by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 4.2-1, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4. The results of applying this process are discussed after Table 5.1-1.

Table 5.1-1
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (BRAIDWOOD BASE CASE)

EPRI/NEI Class	Braidwood Release Category	Unit 1 Frequency Based on Categorized Results (per yr)	Unit 2 Frequency Based on Categorized Results (per yr)
8	A	3.43E-06	3.35E-06
2	B_1	4.91E-08	4.23E-08
8	B_2	8.42E-08	7.91E-08
7	B_3	1.19E-07	2.29E-07
2	C_1	3.23E-10	3.23E-10
8	C_2	2.91E-08	2.91E-08
7	C_3	0	0
2	D	5.83E-07	6.47E-07
8	E	5.42E-09	2.11E-10
2	F_1	4.12E-09	6.64E-09
7	F_2	6.80E-07	1.36E-06
2	G	6.06E-09	6.06E-09
Total LERF		4.99E-06	5.75E-06
Total non-LERF		4.97E-05	4.81E-05
Total CDF		5.46E-05	5.38E-05

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially assigned from the Level 2 Release Categories not listed in Table 5.1-1, minus the EPRI/NEI Class 3a and 3b frequency, calculated below. For simplicity, and since Braidwood only tracks LERF versus non-LERF release categories, all non-LERF endstates are assumed to be in this bin. This may tend to under-predict the calculated plant-specific total dose and CCFP values, but will not change the calculated changes in population dose, LERF, and CCFP that are the determined in this risk impact assessment.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Braidwood Release Categories B_1, C_1, D, F_1, and G, listed in Table 5.1-1.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small ($2L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are determined as follows:

$PROB_{Class_3a}$ = probability of small pre-existing containment liner leakage
= 0.027 [see Section 4.3]
$PROB_{Class_3b}$ = probability of large pre-existing containment liner leakage
= 0.0027 [see Section 4.3]

As described in section 4.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions), or that would include containment spray operation such that a Large Release would be unlikely (currently not credited in this assessment).

For Unit 1:

Class_3a = 0.027 * (CDF-Class 2-Class 8)
= 0.027 * (5.46E-05 – 6.42E-07 – 3.55E-06) = 1.36E-6/yr

Class_3b = 0.0027 * (CDF-Class 2-Class 8)
= 0.0027 * (5.46E-05 – 6.42E-07 – 3.55E-06) = 1.36E-7/yr

For Unit 2:

Class_3a = 0.027 * (CDF-Class 2-Class 8)
= 0.027 * (5.38E-05 – 7.02E-07 – 3.45E-06) = 1.34E-6/yr

Class_3b = 0.0027 * (CDF-Class 2-Class 8)
= 0.0027 * (5.38E-05 – 7.02E-07 – 3.45E-06) = 1.34E-7/yr

For this analysis, the associated containment leakage for Class 3a is $10L_a$ and for Class 3b is $35L_a$. These assignments are consistent with the NEI Interim Guidance.

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the NEI Interim Guidance, however, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined from Release Categories B_3, C_3 and F_2 from the Braidwood Level 2 results.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency is determined from Release Categories A, B_2, C_2 and E from the Braidwood Level 2 results.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definitions of accident classes defined in EPRI-TR-104285 and the NEI Interim Guidance. Table 5.1-2 summarizes these accident frequencies by accident class for Braidwood.

Table 5.1-2
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (BRAIDWOOD BASE CASE)

Accident Classes (Containment Release Type)	Accident Progression Bin (APB)	Description	Unit 1 Frequency (per Rx-yr)		Unit 2 Frequency (per Rx-yr)	
			NEI Methodology	NEI Methodology Plus Corrosion	NEI Methodology	NEI Methodology Plus Corrosion
1	6 & 7	No Containment Failure [(1)]	4.82E-05	4.82E-05	4.66E-05	4.66E-05
2	2	Large Isolation Failures (Failure to Close)	6.42E-07	6.42E-07	7.02E-07	7.02E-07
3a	10 La	Small Isolation Failures (liner breach)	1.36E-06	1.36E-06	1.34E-06	1.34E-06
3b	35 La	Large Isolation Failures (liner breach)	1.36E-07	1.37E-07	1.34E-07	1.35E-07
4	NA	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A
5	NA	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A
6	NA	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A
7	4	Failures Induced by Phenomena (Early and Late)	7.99E-07	7.99E-07	1.59E-06	1.59E-06
8	5	Bypass (SGTR and ISLOCA)	3.55E-06	3.55E-06	3.45E-06	3.45E-06
CDF		All CET end states	5.46E-05	5.46E-05	5.38E-05	5.38E-05

(1) For simplicity, includes all non-LERF endstates for Braidwood. This may tend to under-predict the calculated plant-specific total dose and CCFP values, but will not change the calculated changes in population dose, LERF, and CCFP that are the determined in this risk impact assessment.

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

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences compared to the reference plant as described in Section 4.2, and summarized in Table 4.2-4. The results of applying these releases to the EPRI/NEI containment failure classification are as follows:

Class 1	=	2.15E+03 person-rem (at $1.0L_a$) = 2.15E+03 person-rem [(1)]
Class 2	=	2.79E+06[(2)]
Class 3a	=	2.15E+03 person-rem x $10L_a$ = 2.15E+04 person-rem [(3)]
Class 3b	=	2.15E+03 person-rem x $35L_a$ = 7.53E+04 person-rem [(3)]
Class 4	=	Not analyzed
Class 5	=	Not analyzed
Class 6	=	Not analyzed
Class 7	=	2.13E+06 person-rem [(4)]
Class 8	=	3.50E+06 person-rem [(5)]

(1) The derivation is described in Section 4.2 for Braidwood. Class 1 is assigned the dose from the "no containment failure" APBs from NUREG/CR-4551 (i.e., APB #6 and APB #7). The dose is calculated as an average of the dose for these bins from Table 4.2-2.

(2) The Class 2, containment isolation failures, dose is assigned from APB #2 (Early CF) from Table 4.2-2.

(3) The Class 3a and 3b dose are related to the leakage rate as shown. This is consistent with the NEI Interim Guidance.

(4) The Class 7 dose is assigned from APB #4 (Late CF) from Table 4.2-2.

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from APB #5 (Bypass) from Table 4.2-2.

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [2] containment failure classifications, and consistent with the NEI guidance [3] are provided in Table 5.2-1.

Table 5.2-1

BRAIDWOOD POPULATION DOSE ESTIMATES

FOR POPULATION WITHIN 50 MILES

Accident Classes (Containment Release Type)	Accident Progression Bin (APB)	Description	Person-Rem (50 miles)
1	6 & 7	No Containment Failure	2.15E+03
2	2	Large Isolation Failures (Failure to Close)	2.79E+06
3a	$10L_a$	Small Isolation Failures (liner breach)	2.15E+04
3b	$35L_a$	Large Isolation Failures (liner breach)	7.53E+04
4	NA	Small Isolation Failures (Failure to seal –Type B)	NA
5	NA	Small Isolation Failures (Failure to seal—Type C)	NA
6	NA	Other Isolation Failures (e.g., dependent failures)	NA
7	4	Failures Induced by Phenomena (Early and Late)	2.13E+06
8	5	Bypass (SGTR and Interfacing System LOCA)	3.50E+06

The above dose estimates, when combined with the results presented in Table 5.1-2, yield the Braidwood baseline mean consequence measures for each accident class. These results are presented in Table 5.2-2 for Unit 1 and Table 5.2-3 for Unit 2.

Table 5.2-2

BRAIDWOOD UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.15E+03	4.82E-05	1.04E-01	4.82E-05	1.04E-01	-9.63E-07
2	Large Isolation Failures (Failure to Close)	2.79E+06	6.42E-07	1.79	6.42E-07	1.79	--
3a	Small Isolation Failures (liner breach)	2.15E+04	1.36E-06	2.93E-02	1.36E-06	2.93E-02	--
3b	Large Isolation Failures (liner breach)	7.53E+04	1.36E-07	1.03E-02	1.37E-07	1.03E-02	3.37E-05
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	2.13E+06	7.99E-07	1.70	7.99E-07	1.70	--
8	Bypass (SGTR and ISLOCA)	3.50E+06	3.55E-06	1.24E+01	3.55E-06	1.24E+01	--
CDF	All CET end states		5.46E-05	16.05	5.46E-05	16.05	3.27E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.

2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.2-3

BRAIDWOOD UNIT 2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.15E+03	4.66E-05	1.00E-01	4.66E-05	1.00E-01	-9.47E-07
2	Large Isolation Failures (Failure to Close)	2.79E+06	7.02E-07	1.96	7.02E-07	1.96	--
3a	Small Isolation Failures (liner breach)	2.15E+04	1.34E-06	2.88E-02	1.34E-06	2.88E-02	--
3b	Large Isolation Failures (liner breach)	7.53E+04	1.34E-07	1.01E-02	1.35E-07	1.01E-02	3.32E-05
4	Small Isolation Failures (Failure to seal –Type B)	N/A	NA	NA	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	N/A	NA	NA	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	N/A	NA	NA	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	2.13E+06	1.59E-06	3.39	1.59E-06	3.39	--
8	Bypass (SGTR and ISLOCA)	3.50E+06	3.45E-06	1.21E+01	3.45E-06	1.21E+01	--
CDF	All CET end states		5.38E-05	17.58	5.38E-05	17.58	3.22E-05

1) Only release Classes 1 and 3b are affected by the corrosion analysis.

2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

As indicated previously, the total dose may be slightly under-predicted due to the treatment of all non-LERF endstates as being assigned to EPRI Class 1. Although the total dose may be under-predicted, the Braidwood dose would still compare favorably with other locations given the relative population densities surrounding each location:

Plant	Annual Dose (Person-Rem/Yr)	Reference
Indian Point 3	14,515	[6]
Peach Bottom	6.2	[23]
Crystal River	1.4	[20]
Braidwood Unit 1	15.92	[Table 5.2-2]
Braidwood Unit 2	17.45	[Table 5.2-3]

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

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to fifteen-years. To do this, an evaluation must first be made of the risk associated with the ten-year interval since the base case applies to a 3-year interval (i.e., a simplified representation of a 3-in-10 interval).

Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the NEI guidance as described in Section 4.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 5.3-1 for Unit 1 and Table 5.3-2 for Unit 2.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value, as described in Section 4.3. The results for this calculation are presented in Table 5.3-3 for Unit 1 and Table 5.3-4 for Unit 2.

Table 5.3-1

BRAIDWOOD UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.15E+03	4.47E-05	9.60E-02	4.47E-05	9.60E-02	-5.51E-06
2	Large Isolation Failures (Failure to Close)	2.79E+06	6.42E-07	1.79	6.42E-07	1.79	--
3a	Small Isolation Failures (liner breach)	2.15E+04	4.54E-06	9.75E-02	4.54E-06	9.75E-02	--
3b	Large Isolation Failures (liner breach)	7.53E+04	4.54E-07	3.41E-02	4.56E-07	3.43E-02	1.93E-04
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	2.13E+06	7.99E-07	1.70	7.99E-07	1.70	--
8	Bypass (SGTR and ISLOCA)	3.50E+06	3.55E-06	1.24E+01	3.55E-06	1.24E+01	--
CDF	All CET end states		5.46E-05	16.13	5.46E-05	16.13	1.87E-04

[1] Only release classes 1 and 3b are affected by the corrosion analysis.

[2] Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.3-2

BRAIDWOOD UNIT 2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.15E+03	4.31E-05	9.28E-02	4.31E-05	9.28E-02	-5.42E-06
2	Large Isolation Failures (Failure to Close)	2.79E+06	7.02E-07	1.96	7.02E-07	1.96	--
3a	Small Isolation Failures (liner breach)	2.15E+04	4.46E-06	9.60E-02	4.46E-06	9.60E-02	--
3b	Large Isolation Failures (liner breach)	7.53E+04	4.46E-07	3.36E-02	4.49E-07	3.38E-02	1.90E-04
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	2.13E+06	1.59E-06	3.39	1.59E-06	3.39	--
8	Bypass (SGTR and ISLOCA)	3.50E+06	3.45E-06	1.21E+01	3.45E-06	1.21E+01	--
CDF	All CET end states		5.38E-05	17.66	5.38E-05	17.66	1.84E-04

[1] Only release classes 1 and 3b are affected by the corrosion analysis.

[2] Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.3-3

BRAIDWOOD UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.15E+03	4.22E-05	9.07E-02	4.22E-05	9.06E-02	-1.27E-05
2	Large Isolation Failures (Failure to Close)	2.79E+06	6.42E-07	1.79	6.42E-07	1.79	--
3a	Small Isolation Failures (liner breach)	2.15E+04	6.81E-06	1.46E-01	6.81E-06	1.46E-01	--
3b	Large Isolation Failures (liner breach)	7.53E+04	6.81E-07	5.13E-02	6.87E-07	5.17E-02	4.46E-04
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	2.13E+06	7.99E-07	1.70	7.99E-07	1.70	--
8	Bypass (SGTR and ISLOCA)	3.50E+06	3.55E-06	1.24E+01	3.55E-06	1.24E+01	--
CDF	All CET end states		5.46E-05	16.19	5.46E-05	16.19	4.33E-04

(1) Only release classes 1 and 3b are affected by the corrosion analysis.

(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.3-4

BRAIDWOOD UNIT 2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	2.15E+03	4.07E-05	8.75E-02	4.07E-05	8.75E-02	-1.25E-05
2	Large Isolation Failures (Failure to Close)	2.79E+06	7.02E-07	1.96	7.02E-07	1.96	--
3a	Small Isolation Failures (liner breach)	2.15E+04	6.70E-06	1.44E-01	6.70E-06	1.44E-01	--
3b	Large Isolation Failures (liner breach)	7.53E+04	6.70E-07	5.04E-02	6.76E-07	5.09E-02	4.39E-04
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal— Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	2.13E+06	1.59E-06	3.39	1.59E-06	3.39	--
8	Bypass (SGTR and ISLOCA)	3.50E+06	3.45E-06	1.21E+01	3.45E-06	1.21E+01	--
CDF	All CET end states		5.38E-05	17.72	5.38E-05	17.72	4.26E-04

(1) Only release classes 1 and 3b are affected by the corrosion analysis.

(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.4 Step 4 – Determine the Change in Risk in Terms of Large Early Release Frequency

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the NEI guidance, 100% of the Class 3b contribution would be considered LERF. For Braidwood, however, the Class 3b radionuclide release person-rem is significantly less than a typical LERF contributor as can be seen by comparing the relative population dose for Class 3b to that of Class 2 (7.53E+04 person-rem / 2.79E+06 person-rem or 2.70%). Additionally, as was noted in Section 4.3, a substantial portion of this increase could potentially be non-LERF contributors if the availability of containment sprays were factored into the analysis. As such, based on these two considerations, it should be recognized that classifying all of the Class 3b contributions as LERF is very conservative.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Braidwood, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the NEI guidance methodology). Based on the original 3/10 year test interval assessment from Tables 5.2-2 and 5.2-3, the Class 3b frequency is 1.36E-07/yr for Unit 1 and 1.34E-07/yr for Unit 2. Based on a ten-year test interval from Tables 5.3-1 and 5.3-2, the Class 3b frequency is 4.54E-07/yr for Unit 1 and 4.46E-07/yr for Unit 2; and, based on a fifteen-year test interval from Tables 5.3-3 and 5.3-4, it is 6.81E-07/yr for Unit 1 and

6.70E-07/yr for Unit 2. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 5.45E-07/yr for Unit 1 and 5.36E-07/yr for Unit 2. Similarly, the increase due to increasing the interval from 10 to 15 years is 2.27E-07/yr for Unit 1 and 2.24E-07/yr for Unit 2. As can be seen, even with the conservatisms included in the evaluation (per the NEI methodology), the estimated change in LERF is below the threshold criteria for a small change in risk when comparing the 15 year results to the current 10-year requirement or to the original 3-in-10 year requirement.

5.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability

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

The change in CCFP can be calculated by using the method specified in the NEI Interim Guidance. The NRC has previously accepted similar calculations [7] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy.

Unit	CCFP 3 in 10 yrs	CCFP 1 in 10 yrs	CCFP 1 in 15 yrs	$\Delta CCFP_{15-3}$	$\Delta CCFP_{15-10}$
1	9.38%	9.96%	10.37%	1.00%	0.42%
2	10.93%	11.51%	11.93%	1.00%	0.42%

CCFP = [1 - (Class 1 frequency + Class 3a frequency) / CDF] * 100%

The change in CCFP of approximately 1% by extending the test interval to 15 years from the original 3-in-10 year requirement is judged to be insignificant.

5.6 Summary of Results

The results from this ILRT extension risk assessment for Braidwood are summarized in Table 5.6-1 for Unit 1 and Table 5.6-2 for Unit 2.

Table 5.6-1

Braidwood Unit 1 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.15E+03	4.82E-05	1.04E-01	4.47E-05	9.60E-02	4.22E-05	9.60E-02
2	2.79E+06	6.42E-07	1.79	6.42E-07	1.79	6.42E-07	1.79
3a	2.15E+04	1.36E-06	2.93E-02	4.54E-06	9.75E-02	6.81E-06	1.46E-01
3b	7.53E+04	1.36E-07	1.03E-02	4.56E-07	3.43E-02	6.87E-07	5.17E-02
7	2.13E+06	7.99E-07	1.70	7.99E-07	1.70	7.99E-07	1.70
8	3.50E+06	3.55E-06	1.24E+01	3.55E-06	1.24E+01	3.55E-06	1.24E+01
Total		5.46E-05	16.05	5.46E-05	16.13	5.46E-05	16.19
ILRT Dose Rate from 3a and 3b		3.95E-02		1.32E-01		1.98E-01	
Delta Total Dose Rate	From 3 yr	---		9.21E-02		1.58E-01	
	From 10 yr	---		---		6.60E-02	
3b Frequency (LERF)		1.36E-07		4.54E-07		6.81E-07	
Delta LERF	From 3 yr	---		3.17E-07		5.45E-07	
	From 10 yr	---		---		2.28E-07	
CCFP %		9.38%		9.96%		10.37%	
Delta CCFP %	From 3 yr	---		0.58%		1.00%	
	From 10 yr	---		---		0.42%	

Table 5.6-2

Braidwood Unit 2 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.15E+03	4.66E-05	1.00E-01	4.31E-05	9.28E-02	4.07E-05	8.75E-02
2	2.79E+06	7.02E-07	1.96	7.02E-07	1.96	7.02E-07	1.96
3a	2.15E+04	1.34E-06	2.88E-02	4.46E-06	9.60E-02	6.70E-06	1.44E-01
3b	7.53E+04	1.35E-07	1.01E-02	4.49E-07	3.38E-02	6.76E-07	5.09E-02
7	2.13E+06	1.59E-06	3.39	1.59E-06	3.39	1.59E-06	3.39
8	3.50E+06	3.45E-06	1.21E+01	3.45E-06	1.21E+01	3.45E-06	1.21E+01
Total		5.38E-05	17.58	5.38E-05	17.66	5.38E-05	17.72
ILRT Dose Rate from 3a and 3b		3.89E-02		1.30E-01		1.95E-01	
Delta Total Dose Rate	From 3 yr	---		9.07E-02		1.56E-01	
	From 10 yr	---		---		6.50E-02	
3b Frequency (LERF)		1.34E-07		4.46E-07		6.70E-07	
Delta LERF	From 3 yr	---		3.12E-07		5.36E-07	
	From 10 yr	---		---		2.24E-07	
CCFP %		10.93%		11.51%		11.93%	
Delta CCFP %	From 3 yr	---		0.58%		1.00%	
	From 10 yr	---		---		0.42%	

6.0 SENSITIVITIES

6.1 Sensitivity to Corrosion Impact Assumptions

The results in Tables 5.6-1, 5.6-2 and 6.1-1 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the cylinder and dome and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6.1-1. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 1.75E-7/yr for Unit 1 and 1.73E-7/yr for Unit 2. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6.1-1
Steel Liner Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)	Unit 1 Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 years (per Rx-yr)		Unit 2 Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 years (per Rx-yr)	
			Total Increase	Increase Due to Corrosion	Total Increase	Increase Due to Corrosion
Base Case Doubles every 5 yrs	Base Case (1% Cylinder, 0.1% Basemat)	Base Case 10%	5.50E-07	5.48E-09	5.42E-07	5.39E-09
Doubles every 2 yrs	Base	Base	5.57E-07	1.25E-08	5.49E-07	1.23E-08
Doubles every 10 yrs	Base	Base	5.50E-07	4.62E-09	5.41E-07	4.55E-09
Base	Base	15%	5.53E-07	7.67E-09	5.44E-07	7.55E-09
Base	Base	5%	5.48E-07	3.29E-09	5.39E-07	3.24E-09
Base	10% Cylinder, 1% Basemat	Base	6.00E-07	5.48E-08	5.90E-07	5.39E-08
Base	0.1% Cylinder, 0.01% Basemat	Base	5.45E-07	5.48E-10	5.37E-07	5.39E-10
Lower Bound						
Doubles every 10 yrs	0.1% Cylinder, 0.01% Basemat	5% 100%	5.45E-07	2.77E-10	5.37E-07	2.73E-10
Upper Bound						
Doubles every 2 yrs	10% Cylinder, 1% Basemat	15% 100%	7.20E-07	1.75E-07	7.09E-07	1.73E-07

6.2 EPRI Expert Elicitation Sensitivity

An expert elicitation was performed to reduce excess conservatisms in the data associated with the probability of undetected leak within containment [22]. Since the risk impact assessment of the extensions to the ILRT interval is sensitive to both the probability of the leakage as well as the magnitude, it was decided to perform the expert elicitation in a manner to solicit the probability of leakage as function of leakage magnitude. In addition, the elicitation was performed for a range of failure modes which allowed experts to account for the range of mechanisms of failure, the potential for undiscovered mechanisms, un-inspectable areas of the containment as well as the potential for detection by alternate means. 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 basic difference in the application of the ILRT interval methodology using the expert elicitation is a change in the probability of pre-existing leakage in the containment. The basic methodology uses the Jeffery's 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., $10L_a$ for small and $35L_a$ for large) are used here. Table 6.2-1 illustrates the magnitudes and probabilities of a pre-existing leak in containment associated with the Jeffery's non-informative prior and the expert elicitation statistical treatments. These values are used in the ILRT interval extension for the base methodology and in this sensitivity case. Details of the expert elicitation process, the input to expert elicitation as well as the results of the expert elicitation are available in the various appendices of the EPRI report [22].

Table 6.2-1
EPRI Expert Elicitation Results

Leakage Size (La)	Jeffery's Non-Informative Prior	Expert Elicitation Mean Probability of Occurrence	Percent Reduction
10	2.7E-02	3.88E-03	86%
35	2.7E-03	9.86E-04	63%

A summary of the results using the expert elicitation values for probability of containment leakage is provided in Table 6.2-2 for Unit 1 and Table 6.2-3 for Unit 2. As mentioned previously, probability values are those associated with the magnitude of the leakage used in the Jeffery's non-informative prior evaluation (10 L_a for small and 35 L_a for large). The expert elicitation process produces a probability versus leakage magnitude relationship and it is possible to assess higher leakage magnitudes more reflective of large early releases but these evaluations are not performed in this study. Alternative leakage magnitudes could include consideration of 100 – to 600 L_a where leakage begins to approach large early releases.

The net affect is that the reduction in the multipliers shown above has the same impact on the calculated increases in the LERF values. The increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 1.99E-07/yr for Unit 1 and 1.96E-07/yr for Unit 2. Similarly, the increase due to increasing the interval from 10 to 15 years is 8.31E-08/yr for Unit 1 and 8.18E-08/yr for Unit 2. As such, if the expert elicitation mean probability of occurrences are used instead of the non-informative prior estimates, the change in LERF for Braidwood is below the threshold criteria for a "very small" change in risk when comparing the 15 year results to the current 10-year requirement and is just above the "very small" change threshold value of 1.0E-7/yr in the "small" change region when compared to the original 3-in-10 year requirement. The results of this sensitivity study are

judged to be more indicative of the actual risk associated with the ILRT extension than the results from the assessment as dictated by the NEI methodology values, and yet are still conservative given the assumption that all of the Class 3b contribution is considered to be LERF.

Table 6.2-2

Braidwood Unit 1 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Based on EPRI Expert Elicitation Leakage Probabilities)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.15E+03	4.94E-05	1.06E-01	4.88E-05	1.05E-01	4.84E-05	1.04E-01
2	2.79E+06	6.42E-07	1.79	6.42E-07	1.79	6.42E-07	1.79
3a	2.15E+04	1.96E-07	4.21E-03	6.52E-07	1.40E-02	9.79E-07	2.10E-02
3b	7.53E+04	4.98E-08	3.75E-03	1.66E-07	1.25E-02	2.49E-07	1.87E-02
7	2.13E+06	7.99E-07	1.70	7.99E-07	1.70	7.99E-07	1.70
8	3.50E+06	3.55E-06	1.24E+01	3.55E-06	1.24E+01	3.55E-06	1.24E+01
Total		5.46E-05	16.02	5.46E-05	16.04	5.46E-05	16.05
ILRT Dose Rate from 3a and 3b		7.96E-03		2.65E-02		3.98E-02	
Delta Total Dose Rate	From 3 yr	---		1.85E-02		3.18E-02	
	From 10 yr	---		---		1.33E-02	
3b Frequency (LERF)		4.98E-08		1.66E-07		2.49E-07	
Delta LERF	From 3 yr	---		1.16E-07		1.99E-07	
	From 10 yr	---		---		8.31E-08	
CCFP %		9.22%		9.43%		9.58%	
Delta CCFP %	From 3 yr	---		0.21%		0.36%	
	From 10 yr	---		---		0.15%	

Table 6.2-3

Braidwood Unit 2 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions ((Based on EPRI Expert Elicitation Leakage Probabilities)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	2.15E+03	4.87E-05	1.05E-01	4.81E-05	1.03E-01	4.77E-05	1.03E-01
2	2.79E+06	7.02E-07	1.96	7.02E-07	1.96	7.02E-07	1.96
3a	2.15E+04	1.93E-07	4.14E-03	6.42E-07	1.38E-02	9.63E-07	2.07E-02
3b	7.53E+04	4.90E-08	3.69E-03	1.63E-07	1.23E-02	2.45E-07	1.84E-02
7	2.13E+06	1.59E-06	3.39	1.59E-06	3.39	1.59E-06	3.39
8	3.50E+06	3.45E-06	1.21E+01	3.45E-06	1.21E+01	3.45E-06	1.21E+01
Total		5.46E-05	17.55	5.46E-05	17.57	5.46E-05	17.58
ILRT Dose Rate from 3a and 3b		7.83E-03		2.61E-02		3.91E-02	
Delta Total Dose Rate	From 3 yr	---		1.82E-02		3.13E-02	
	From 10 yr	---		---		1.31E-02	
3b Frequency (LERF)		4.90E-08		1.63E-07		2.45E-07	
Delta LERF	From 3 yr	---		1.14E-07		1.96E-07	
	From 10 yr	---		---		8.18E-08	
CCFP %		10.61%		10.82%		10.97%	
Delta CCFP %	From 3 yr	---		0.21%		0.36%	
	From 10 yr	---		---		0.15%	

6.3 Potential Impact from External Events Contribution

In the Braidwood IPEEE, the dominant risk contributor from external events was found to be from fire events. Other potential contributors such as seismic and high winds were found to be within acceptable limits. As a reasonable assessment of the impact from external events, one can assume that the external events CDF is comparable to the internal events CDF. Additionally, one can assume that the fractional LERF contribution from the internal events model (excluding the contribution from ISLOCA or SGTR scenarios since these types of events would typically not occur from an external event initiator) also provides a reasonable estimate of the LERF impact from external events.

For Braidwood Unit 1, the reported total Internal Events LERF as determined from a simplified LERF model is 4.99E-06/yr, and for Unit 2 it is 5.75E-06/yr [16]. As indicated above, the External Events baseline LERF would be expected to be less than the Internal Events baseline LERF because some of the Internal Events baseline LERF comes from events that are not events that are initiated by fires (i.e., ISLOCA and SGTR). Subtracting off the contributions from these events (i.e. EPRI Class 8) of 3.55E-6/yr for Unit 1 and 3.45E-6/yr for Unit 2 yields a LERF value subject to the External Events impact of 1.44E-6/yr for Unit 1 and 2.30E-6/yr for Unit 2. There are some known conservatisms in the simplified LERF model, but these values will be used in the discussion below for illustration purposes.

However, as is shown in Table 6.3-1, if it is assumed that the LERF impact of the ILRT extension from External Events is assumed to be the same as that from Internal Events, the total LERF would be below the Regulatory Guide 1.174 criteria of 1.0E-05 following the ILRT extension. Using the same assumptions, Table 6.3-2 shows the impact if the EPRI expert elicitation values are used for the Class 3a and 3b frequency determination. In this case, the total LERF is further below the Regulatory Guide 1.174 criteria of 1.0E-05 following the ILRT extension.

Table 6.3-1
Braidwood Estimated Total LERF Including External Events Impact
(Base Case NEI Methodology)

Contributor	Braidwood Unit 1	Braidwood Unit 2
Internal Events LERF	4.99E-06	5.75E-06
External Events LERF	1.44E-06	2.30E-06
Internal Events LERF due to ILRT (at 15 years)	6.81E-07	6.70E-07
External Events LERF due to ILRT (at 15 years)	6.81E-07	6.70E-07
Total:	**7.79E-06**	**9.39E-06**

Table 6.3-2
Braidwood Estimated Total LERF Including External Events Impact
(EPRI Expert Elicitation Methodology)

Contributor	Braidwood Unit 1	Braidwood Unit 2
Internal Events LERF	4.99E-06	5.75E-06
External Events LERF	1.44E-06	2.30E-06
Internal Events LERF due to ILRT (at 15 years)	2.49E-07	2.45E-07
External Events LERF due to ILRT (at 15 years)	2.49E-07	2.45E-07
Total:	**6.93E-06**	**8.54E-06**

7.0 CONCLUSIONS

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

- Reg. Guide 1.174 [4] 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 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 internal events LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is very conservatively estimated as 5.45E-7/yr for Unit 1 and 5.36E-7/yr for Unit 2 using the NEI guidance as written, and at 1.99E-07/yr for Unit 1 and 1.96E-07/yr for Unit 2 using the EPRI Expert Elicitation methodology. These values could also be reduced if the potential impact from the availability of containment sprays were factored into the analysis, but even without accounting for this reduction, the estimated change in LERF is determined to be "small" using the acceptance guidelines of Reg. Guide 1.174.
- Regulatory Guide 1.174 [4] also states that when the calculated increase in LERF is in the range of 1.0E-06 per reactor year to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. As such, an additional assessment of the impact from external events was also made. In that case, the total LERF was conservatively estimated as 7.79.E-06/yr and 9.39E-06/yr for Braidwood Units 1 and 2, respectively using the NEI guidance directly. These numbers fall to 6.93E-6/yr and 8.54E-6/yr if the EPRI Expert Elicitation methodology is utilized. These values are all below the RG 1.174 acceptance criteria for total LERF of 1.0E-05, but the EPRI Expert Elicitation

methodology provides more margin to the limit than that provided by the NEI methodology directly.

- The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.15 person-rem/yr for Unit 1 and 0.14 person-rem/yr for Unit 2 using the NEI guidance, and drops to 0.03 person-rem/yr for both Units using the EPRI Expert Elicitation methodology. Therefore, in either case, the risk impact when compared to other severe accident risks is negligible.
- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen year interval is about 1% using the NEI guidance, and drops to about 0.4% using the EPRI Expert Elicitation methodology. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the Braidwood Station risk profile.

Previous Assessments

The NRC in NUREG-1493 [5] has previously concluded that:

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

containment penetrations, ILRTs also test the integrity of the containment structure.

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

8.0 REFERENCES

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Appendix A CDF and LERF Subcategory Calculations

CDF is available from Rev. 5E Quantification Notebook [Ref. 1].

LERF Sequence frequencies are calculated using the following process:

1. Quantify each sequence using PRAQuant at a truncation limit of 1E-11. Results are stored in separate cutset files (i.e., one sequence per file). This is done so that each cutset is tagged with a "class" label that identifies the sequence. PRAQuant files are listed in Reference 6.
2. Use Merger32.exe [Ref. 4] to combine sequences into one cutset file. This needs to be done in order to subsume non-minimal or duplicate cutsets (i.e., cutsets that satisfy more than one sequence logic).
3. Use CSUTIL32.exe [Ref. 5]"Set Event Flags" feature on the merged cutset file. Set 1.0 events to TRUE and subsume. The result of this action is a cutset file that matches the base model quantification at the same truncation frequency.
4. Use CSUTIL32.exe "Split Classes" feature to split the cutsets into classes (that represent sequences). The results of this action are a cutset file containing all sequences with a separate module for each sequence, and a listing of each class (sequence) with the number of cutsets and total LERF for that sequence. This file is sent to the clipboard by CSUTIL32 and can be pasted into Excel.

After the LERF sequence frequencies are quantified, and the results stored in an Excel spreadsheet, the LERF category frequencies are obtained using the following process:

1. Note that each sequence designator (class) ends with a letter that represents the LERF category associated with that Level 1 sequence. The sequence designator, and hence the LERF category assignment is provided in Table D.3 of the Quantification Notebook (BB PRA-014), Appendix D [Ref. 1]
2. LERF Category B is divided into B_1, B_2 and B_3. The contribution of each sequence to subcategories B_2 and B_3 is determined using the Fussel-Vesely importance of the following basic events, multiplied by the total sequence frequency:

   B_2: ISGTR

   B_3: CF-VB1-U1 or CF-VB1-U2
3. The frequency of LERF Category B_1 is the remainder of the sequence frequency not assigned to B_2 and B_3.

4. LERF Category C is divided into C_1, C_2 and C_3. The contribution of each sequence to these subcategories is determined the same way as the Category B subcategories.
5. LERF Category F is divided into F_1 and F_2. The contribution of each sequence to subcategory F_2 is determined using the Fussel-Vesely importance for ISGTR multiplied by the total sequence frequency. Subcategory F_1 frequency is the remainder of the sequence frequency not assigned to F_2.
6. The sum totals for the category and subcategory frequencies from the Excel spreadsheet [Ref. 2] are copied to Table A-1. These values are also used in Table 4.2-1 where the frequency assignments to the different accident progression bins and EPRI release categories are made to perform the ILRT extension assessment.

Sensitivity for URE BB-591

URE BB-591 relates to missing power supplies for certain containment isolation valves in the Unit 2 model. If the model were corrected to accurately reflect the power supplies for these valves, the calculations performed for the ILRT extension would be affected. Therefore, a sensitivity analysis was performed to determine the impact of this model error on the ILRT calculations.

1. The Revision 5E fault tree (Master5E.caf) [Ref. 1] was modified to include the following dependencies:

Valve	Bus
2SI8814	MCC 231X1A
2SI8920	MCC 231X1A
2SI8813	MCC 232X4A

2. The model quantifications and manipulations described in the previous section were repeated for Unit 2.
3. The calculations performed by the spreadsheet, as described in the previous section, were performed for the new LERF results. This includes updating the Fussell-Vesely importance values for ISGTR and Containment Failure at Vessel Breach, for those sequences with frequency changes as a result of the sensitivity.
4. The results of the new LERF category calculations [Ref. 3] are shown in Table A-1. As can be seen, there are insignificant changes to a few of the LERF subcategories. Therefore, the results of the original calculations would not be significantly impacted and the conclusions remain valid.

Table A-1 Braidwood LERF Model Subcategory Development

Braidwood LERF Category	Definition	Unit 1 Frequency/ yr	Unit 2 Frequency /yr	Unit 2 Sensitivity Frequency	Unit 2 Sensitivity Delta
A	Straight pass through CDF sequence to LERF	3.43E-06	3.35E-06	3.35E-06	0
B_1	High pressure sequences with no AFW available.	4.91E-08	4.23E-08	4.50E-08	3E-09
B_2	High pressure sequences with no AFW available, where the possibility exists for an ISGTR.	8.42E-08	7.91E-08	7.94E-08	3E-10
B_3	High pressure sequences with no AFW available and questions whether the containment fails at the time of vessel breach.	1.19E-07	2.29E-07	2.30E-07	1E-09
C_1	High pressure sequences with no AFW available.	3.23E-10	3.23E-10	3.23E-10	0
C_2	High pressure sequences with no AFW available, where the possibility exists for an ISGTR.	2.91E-08	2.91E-08	2.91E-08	0
C_3	High pressure sequences with no AFW available and the time between core uncovery and vessel breach is greater than the required evacuation time.	0	0	0	0
D	Sequences that do not lead to containment failure or result in containment failure many hours after core uncovery. Containment isolation is asked.	5.83E-07	6.47E-07	6.52E-07	5E-09
E	SGTR sequences where isolation of the ruptured SG is possible but has not been questioned in the Level 1 event tree, and asks to see isolation of the ruptured SG is successful.	5.42E-09	2.11E-10	2.11E-10	0
F_1	High pressure sequences with AFW available.	4.12E-09	6.64E-09	9.10E-09	2E-09
F_2	High pressure sequences with AFW available, where no possibility exists for an ISGTR.	6.80E-07	1.36E-06	1.35E-06	-1E-08
G	Sequences where the RCS pressure is low at the time of vessel breach and AFW available, where no possibility exists for an ISGTR.	6.06E-09	6.06E-09	6.06E-09	0

Appendix A References

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[6] PRAQuant Files

A11-5E-sCDF.qnt	46 KB	8/8/2006 6:42 pm
A11-5E-sLERF.qnt	46 KB	8/8/2006 7:36 pm
A22-5E-sCDF.qnt	46 KB	8/9/2006 2:22 pm
A22-5E-sLERF.qnt	46 KB	8/9/2006 9:44 am

ATTACHMENT 5

Risk Assessment for Byron Unit 1 and Unit 2 to Support ILRT (Type A) Interval Extension Request

RM DOCUMENTATION NO.	BB PRA-017.54B	REV: 3	PAGE NO. 1

STATION: BYRON

UNIT(S) AFFECTED: 1 and 2

TITLE: Risk Assessment for Byron Unit 1 and Unit 2 To Support ILRT (Type A) Interval Extension Request

SUMMARY (Include UREs incorporated): The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Byron Unit 1 and Byron Unit 2 containment Type A integrated leak rate test (ILRT) interval from 10 years to 15 years.

Revision 3 removed incorporation of specific ILRT test results and changed verb tense in background section.

Internal RM Documentation

Electronic Calculation Data Files:
(Program Name, Version, File Name extension/size/date/hour/min)

Refer to Appendix A of the document.

Main Body

Prepared by: Donald E. Vanover / Donald E. Vanover / 3/2/07
Print Sign Date

Reviewed by: Leo B. Shanley / [signature] / 3/2/07
Print Sign Date

Appendix A

Prepared by: Leo B. Shanley / [signature] / 3/2/07
Print Sign Date

Reviewed by Donald E. Vanover / Donald E. Vanover / 3/2/07
Print Sign Date

Method of Review: [X] Detailed [ ] Alternate

This RM documentation supersedes: Rev. 2 **in its entirety.**

Approved by: Gregory A. Krueger / [signature] / 3/5/07
Print Sign Date

External RM Documentation

Reviewed by: N/A / /
Print Sign Date

Approved by: N/A / /
Print Sign Date

Do any ASSUMPTIONS / ENGINEERING JUDGEMENTS require later verification? **[ ] Yes [X] No**
Tracked By: AT#, URE# etc.)

TABLE OF CONTENTS

Section **Page**

1.0 PURPOSE OF ANALYSIS 3
1.1 Purpose 3
1.2 Background 3
1.3 Criteria 5
2.0 METHODOLOGY 6
3.0 GROUND RULES 8
4.0 INPUTS 10
4.1 General Resources Available 10
4.2 Plant-Specific Inputs 18
4.3 Impact of Extension on Detection of Component Failures That Lead to Leakage (Small and Large) 24
4.4 Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage 27
5.0 RESULTS 34
5.1 Step 1 – Quantify the Base-Line Risk in Terms of Frequency per Reactor Year 36
5.2 Step 2 – Develop Plant-Specific Person-Rem Dose (Population Dose) per Reactor Year 41
5.3 Step 3 – Evaluate Risk Impact of Extending Type A Test Interval From 10-to-15 Years 45
5.4 Step 4 – Determine the Change in Risk in Terms of Large Early Release Frequency 51
5.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability 52
5.6 Summary of Results 53
6.0 SENSITIVITIES 56
6.1 Sensitivity to Corrosion Impact Assumptions 56
6.2 EPRI Expert Elicitation Sensitivity 58
6.3 Potential Impact from External Events Contribution 62
7.0 CONCLUSIONS 64
8.0 REFERENCES 67
Appendix A CDF and LERF Subcategory Calculations 70

1.0 PURPOSE OF ANALYSIS

1.1 Purpose

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Byron Station Units 1and 2 containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3, 21], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [19]. The format of this document is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the December 2005 EPRI final report [22].

1.2 Background

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is 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 normal containment leakage of $1.0L_a$ (allowable leakage).

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [5], "Performance-Based Containment Leak Test Program," September 1995, provides the

technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285.

The NRC report, Performance Based Leak Test Program, NUREG-1493 [5], analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis contained herein has been performed to confirm these conclusions based on Byron specific models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In November and December 2001, NEI issued enhanced guidance (hereafter referred to as the NEI Interim Guidance) that builds on the TR-104285 methodology and intended to provide for more consistent submittals [3,21]. The NEI Interim Guidance was developed for NEI by EPRI using personnel who also developed the TR-104285 methodology. This ILRT interval extension risk assessment for Byron employs the NEI Interim Guidance methodology.

It should be noted that, in addition to ILRT tests, containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC

pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment 3 times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency. Type C tests are also not affected by the Type A test frequency change.

1.3 Criteria

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

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter based on the precedent set by previous submittals for ILRT extensions [6, 20, 23]. (No criteria have been established for this parameter change.)

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years [22]. The approach is consistent with that presented in NEI Interim Guidance [3, 21], EPRI TR-104285 [2], NUREG-1493 [5] and the Calvert Cliffs liner corrosion analysis [19]. The analysis uses results from a Level 2 analysis of core damage scenarios from the current Byron PRA model and subsequent containment response resulting in LERF and non-LERF endstates. This risk assessment is applicable to Byron Units 1 and 2.

The six general steps of this assessment are as follows:

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

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore,

- Consistent with the other industry containment leak risk assessments, the Byron assessment uses population dose as one of the risk measures. The other risk measures used in the Byron assessment are LERF and the conditional containment failure probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.
- This evaluation for Byron uses ground rules and methods to calculate changes in risk metrics that are similar to those used in the EPRI approach.

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The Byron Level 1 and LERF internal events PRA models provide representative results.
- It is appropriate to use the Byron internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations.
- Dose results for the containment failures modeled in the PRA can be characterized by information provided in NUREG/CR-4551 [9]. They are estimated by scaling the NUREG/CR-4551 results by population differences for Byron compared to the NUREG/CR-4551 reference plant.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is $1L_a$. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is $10L_a$, based on the previously approved methodology performed for Indian Point Unit 3 [6, 7].
- The representative containment leakage for Class 3b sequences is $35L_a$, based on the previously approved methodology [6, 7].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [6, 7]. The Class 3b category increase is used as a surrogate for LERF in this application even though the releases associated with a $35L_a$ release would not necessarily be consistent with a "Large" release for Byron. (See, for example, the calculated population dose results for EPRI Class 3b in Table 5.2-2 of 1.81E4 person-rem compared to the 8.41E5 person-rem associated with EPRI Class 8 for containment bypass scenarios.)

- For simplicity, and since Byron only tracks LERF versus non-LERF release categories, all non-LERF endstates are assumed to be in EPRI Class 1. This may tend to under-predict the calculated plant-specific total dose and CCFP values, but will not change the calculated changes in population dose, LERF, and CCFP that are the determined in this risk impact assessment.
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The use of estimated 2010 population data is adequate for this analysis. Scaling the year 2010 population data to the date of the next ILRT test if extended beyond the current due date would not significantly impact the quantitative results, nor would it change the conclusions.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8].

4.0 INPUTS

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

4.1 General Resources Available

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

1. NUREG/CR-3539 [10]
2. NUREG/CR-4220 [11]
3. NUREG-1273 [12]
4. NUREG/CR-4330 [13]
5. EPRI TR-105189 [8]
6. NUREG-1493 [5]
7. EPRI TR-104285 [2]
8. NUREG-1150 [14] and NUREG/CR-4551 [9]
9. NEI Interim Guidance [3, 21]
10. Calvert Cliffs liner corrosion analysis [19]
11. EPRI 1009325 [22]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different

containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the bases for the consequence analysis of the ILRT interval extension for Byron. The ninth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the last study complements the previous EPRI report [2], integrates the NEI interim guidance, and provides the results of an expert elicitation process to determine the relationship between pre-existing containment leakage probability and magnitude.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [15] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories (PNL) for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage. The study calculated unavailabilities for Technical Specification leakages and "large" leakages. NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 PWR events in 740 reactor years and conservatively assuming a one-year duration for each event.

NUREG-1273 [12]

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

NUREG/CR-4330 [13]

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

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

EPRI TR-105189 [8]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk.

The result of the study concluded that a small but measurable safety benefit (shutdown CDF reduced by 1E-8/yr to 1E-7/yr) is realized from extending the test interval from 3 per 10 years to 1 per 10 years.

NUREG-1493 [5]

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

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2 – 1.0%) population risk.
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

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

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage sequences into eight categories of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures

7. Containment failure due to core damage accident phenomena
8. Containment bypass

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

> *"These study results show that the proposed CLRT* [containment leak rate tests] *frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year..."*

Release Category Definitions

Table 4.1-1 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI/NEI methodology [2]. These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

Table 4.1-1

EPRI/NEI CONTAINMENT FAILURE CLASSIFICATIONS [2]

Class	Description
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

NUREG-1150 [14] and NUREG/CR 4551 [9]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding the Surry Power Station. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the Byron LERF end-

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

NEI Interim Guidance [3, 21]

NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions of Containment Integrated Leakage Rate Test Surveillance Intervals" [3] has been developed to provide utilities with revised guidance regarding licensing submittals. Additional information from NEI on the "Interim Guidance" was supplied in Reference [21].

A nine step process is defined which includes changes in the following areas of the previous EPRI guidance:

- Impact of extending surveillance intervals on dose
- Method used to calculate the frequencies of leakages detectable only by ILRTs
- Provisions for using NUREG-1150 dose calculations to support the population dose determination.

The guidance provided in this document builds on the EPRI risk impact assessment methodology [2] and the NRC performance-based containment leakage test program [5], and considers approaches utilized in various submittals, including Indian Point 3 [6,7] (and associated NRC SER) and Crystal River [20].

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

Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [19].

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

EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [22]

This report presents a risk impact assessment for extending integrated leak rate test (ILRT) surveillance intervals to 15 years and is consistent in nature with the NEI interim guidance. This risk impact assessment complements the previous EPRI report, TR-104285, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals.* The earlier report considered changes to local leak rate testing intervals as well as changes to ILRT testing intervals. The original risk impact assessment considers the change in risk based on population dose, whereas the revision considers dose as well as large early release frequency (LERF) and conditional containment failure probability (CCFP). This report deals with changes to ILRT testing intervals and is intended to provide bases for supporting changes to industry (NEI) and regulatory (NRC) guidance on ILRT surveillance intervals.

The risk impact assessment using the Jeffery's Non-Informative Prior statistical method is further supplemented with a sensitivity case using expert elicitation performed to address conservatisms. The expert elicitation is used to determine the relationship between pre-existing containment leakage probability and magnitude. The results of the expert elicitation process from this report are used as a separate sensitivity investigation for the Byron analysis presented here in Section 6.2.

4.2 Plant-Specific Inputs

The Byron specific information used to perform this ILRT interval extension risk assessment includes the following:

- Level 1 Model results [16]
- LERF Model results [16]
- Population within a 50-mile radius [18]
- ILRT results to demonstrate adequacy of the administrative and hardware issues (1)

Byron Internal Events Level 1 PRA

The Byron Level 1 PRA used as input to this analysis is characteristic of the as-built, as-operated plant. The current Level 1 PRA model is a linked fault tree model characteristic of the as-built plant. The total internal events core damage frequency (CDF) used in this analysis is 5.78E-05/yr for Unit 1 and 5.73E-05/yr for Unit 2 [16].

Byron Internal Events LERF Model

Table 4.2-1 summarizes the pertinent Byron results in terms of EPRI/NEI accident class and NUREG/CR-4551 accident progression bin. The total internal events large early release frequency (LERF) used in this analysis is 4.72E-06/yr for Unit 1 and 5.62E-06/yr for Unit 2. For simplicity, and since Byron only tracks LERF versus non-LERF release categories, as detailed in Section 5.1 of this report all non-LERF endstates are assumed to be in EPRI Class 1. This may tend to under-predict the calculated plant-specific total dose and CCFP values, but will not change the calculated changes in population dose, LERF, and CCFP that are the determined in this risk impact assessment.

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(1) The two most recent Type A tests at Byron Unit 1 and Unit 2 have been successful, so the current Type A test interval requirement is 10 years [24].

Table 4.2-1 Byron LERF Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins and EPRI / NEI Accident Classes

Byron LERF Category[1]	Definition	NUREG/ CR-4551 APB	EPRI / NEI Class	Unit 1 Frequency /yr	Unit 2 Frequency/ yr
A	Straight pass through CDF sequence to LERF	5	8	3.16E-06	3.12E-06
B_1	High pressure sequences with no AFW available.	2	2	4.22E-08	3.92E-08
B_2	High pressure sequences with no AFW available, where the possibility exists for an ISGTR.	5	8	7.00E-08	7.01E-08
B_3	High pressure sequences with no AFW available and questions whether the containment fails at the time of vessel breach.	2	7	9.84E-08	2.02E-07
C_1	High pressure sequences with no AFW available.	2	2	1.62E-10	1.62E-10
C_2	High pressure sequences with no AFW available, where the possibility exists for an ISGTR.	5	8	2.01E-08	2.01E-08
C_3	High pressure sequences with no AFW available and the time between core uncovery and vessel breach is greater than the required evacuation time.	2	7	0	0
D	Sequences that do not lead to containment failure or result in containment failure many hours after core uncovery. Containment isolation is asked.	2	2	5.29E-07	6.03E-07
E	SGTR sequences where isolation of the ruptured SG is possible but has not been questioned in the Level 1 event tree, and asks to see isolation of the ruptured SG is successful.	5	8	5.42E-09	2.11E-10
F_1	High pressure sequences with AFW available.	2	2	6.72E-09	4.62E-09
F_2	High pressure sequences with AFW available, where no possibility exists for an ISGTR.	2	7	7.79E-07	1.55E-06
G	Sequences where the RCS pressure is low at the time of vessel breach and AFW available, where no possibility exists for an ISGTR.	3	2	6.06E-09	6.06E-09

[1] Byron LERF categories are indicated by the letters. Where a given Byron LERF category comprises more than one NUREG APB or EPRI category, subscripts have been assigned in this study to denote the portions of the LERF categories that correspond to the noted NUREG APBs or EPRI categories. The determination of these frequencies for Byron Unit 1 and Unit 2 for use in this analysis is described in Appendix A.

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for Byron. Each of the release categories from Table 4.2-1 was associated with an applicable Collapsed Accident Progression Bin (APB) from NUREG/CR-4551 (see below). The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 7 bins that are relevant to the analysis. The definitions of the 7 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 4.2-2 for references purposes. Table 4.2-3 then summarizes the calculated population dose for Surry associated with each APB from NUREG/CR-4551.

Table 4.2-2

Summary Accident Progression Bin (APB) Descriptions [9]

Summary APB Number	Summary Accident Progression Bin (APB) Description
1	CD, VB, Early CF, Alpha Mode Core damage occurs followed by a very energetic molten fuel-coolant interaction in the vessel; the vessel fails and generates a missile that fails the containment as well. Includes accidents that have an Alpha mode failure of the vessel and the containment except those that follow Event V or an SGTR. It includes Alpha mode failures that follow isolation failures because the Alpha mode containment failure is of rupture size.
2	CD, VB, Early CF, RCS Pressure > 200psia Core Damage occurs followed by vessel breach. Implies Early CF with the RCS above 200 psia when the vessel fails. Early CF means at or before VB, so it includes isolation failures and seismic containment failures at the start of the accident as well as containment failure at VB. It does not include bins in which containment failure at VB follows Event V or an SGTR, or Alpha mode failures.
3	CD, VB, Early CF, RCS Pressure < 200 psia Core damage occurs followed by vessel breach. Implies Early CF with the RCS below psia when the containment fails. It does not include bins in which the containment fails at VB, an SGTR, or Alpha mode failures.
4	CD, VB, Late CF Core Damage occurs followed by vessel breach. Includes accidents in which the containment was not failed or bypassed before the onset of core-concrete interaction (CCI) and in which the vessel failed. The failure mechanisms are hydrogen combustion during CCI, Basemat Melt-Through (BMT) in several days, or eventual overpressure due to the failure to provide containment heat removal in the days following the accident.
5	CD, Bypass Core Damage occurs followed by vessel breach. Includes Event V and SGTRs no matter what happens to the containment after the start of the accident. It also includes SGTRs that do not result in VB.
6	CD, VB, No CF Core Damage occurs followed by vessel breach. Includes accidents not evaluated in one of the previous bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.
7	CD, No VB Core Damage occurs but is arrested in time to prevent vessel breach. Includes accident progressions that avoid vessel failures except those that bypass the containment. Most of the bins placed in this bin have no containment failure as well as no VB. It also includes bins in which the containment is not isolated at the start of the accident and the core is brought to a safe stable state before the vessel fails.

Table 4.2-3
Calculation of Surry Population Dose Risk at 50 Miles [9]

Collapsed Bin #	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (person-rem/yr, mean) (2)	NUREG/CR-4551 Collapsed Bin Frequencies (per year) (3)	NUREG/CR-4551 Population Dose at 50 miles (person-rem) (4)
1	0.029	0.158	1.23E-07	1.28E+06
2	0.019	0.106	1.64E-07	6.46E+05
3	0.002	0.013	2.01E-08	6.46E+05 (5)
4	0.216	1.199	2.42E-06	4.95E+05
5	0.732	4.060	5.00E-06	8.12E+05
6	0.001	0.006	1.42E-05	4.23E+02
7	0.002	0.011	1.91E-05	5.76E+02
Totals	1.000	5.55	4.1E-05	

(1) Mean Fractional Contribution to Risk calculated from the average of two samples delineated in Table 5.1-3 of NUREG/CR-4551.

(2) The total population dose risk at 50 miles from internal events in person-rem is provided as the average of two samples in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.

(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-3. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.

(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.

(5) Assumed population dose at 50 miles for Collapsed Bin #3 equal to that of Collapsed Bin #2. Collapsed Bin Frequency #3 was then back calculated using that value. This does not influence the results of this evaluation since Bin #3 does not appear as part of the results for Byron.

Population Estimate Methodology

The person-rem results in Table 4.2-3 can be used as an approximation of the dose for Byron if it is corrected for the population surrounding Byron. The total population within a 50-mile radius of Byron is projected to be 1.274E+06 by the year 2010 [18]. This value is slightly less than the projected value of 1.372E+06 from the Byron UFSAR since it factors in more recent actual census data from 1990 and 2000 for the projected growth estimates compared to the earlier population data utilized in the UFSAR. The use of the 2010 estimate is judged to be sufficient to perform this assessment. Scaling the year 2010 population data to the date of the next ILRT test if extended beyond the current due date would not significantly impact the quantitative results, nor would it change the conclusions.

This population value is compared to the population value that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for Byron.

Total Byron $\text{Population}_{50\text{miles}}$ = 1.274E+06
Surry Population from NUREG/CR-4551 = 1.23E+06
Population Dose Factor = 1.274E+06 / 1.23E+06 = 1.036

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. This does not take into account differences in meteorology data, detailed environmental factors or detailed differences in containment designs, but does provide a first-order approximation for Byron of the population doses associated with each of the release categories from NUREG/CR-4551. This is considered adequate since the conclusions from this analysis will not be substantially affected by the actual dose values that are used.

Table 4.2-4 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for Byron.

Table 4.2-4
Calculation of Byron Population Dose Risk at 50 Miles

Accident Progression Bin (APB)	NUREG/CR-4551 Population Dose at 50 miles (person-rem)	Bin Multiplier used to obtain Byron Population Dose	Byron Adjusted Population Dose at 50 miles (person-rem)
1	1.28E+06	1.036	1.33E+06
2	6.46E+05	1.036	6.69E+05
3	6.46E+05	1.036	6.69E+05
4	4.95E+05	1.036	5.13E+05
5	8.12E+05	1.036	8.41E+05
6	4.23E+02	1.036	4.38E+02
7	5.76E+02	1.036	5.97E+02

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the Byron Level 2 model are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in that document, it was necessary to match the Byron Level 2 release categories to the collapsed APBs. The assignments are shown in Table 4.2-1, along with the corresponding EPRI/NEI classes.

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

The ILRT can detect a number of component failures such as liner breach, failure of certain bellows arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 4.1-1 is divided into two sub-classes representing small and large leakage failures. These subclasses are defined as Class 3a and Class 3b, respectively.

The probability of the EPRI Class 3a failures may be determined, consistent with the NEI Guidance [3], as the mean failure estimated from the available data (i.e., 5 "small" failures in 182 tests leads to a 5/182=0.027 mean value). For Class 3b, using the original NEI Guidance [3], a non-informative prior distribution would be assumed for no "large" failures in 182 tests (i.e., 0.5/(182+1) = 0.0027).

In a follow-on letter [21] to their ILRT guidance document [3], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC regulatory guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> "The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage."

The application of this additional guidance to the analysis for Byron, as detailed in Section 5, involves the following:

1. The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the Byron Level 2 PRA analysis.

2. Since the Byron Level 2 model only includes a representation of LERF vs. non-LERF endstates, for simplification all of the non-LERF endstates have been assigned to EPRI Class 1, and as such are included in the CDF multiplier that is subject to the potential impacts from the ILRT extension assessment. In fact, however, a review of Class 1 accident sequences shows that several of these cases could involve successful operation of containment sprays in which case the potential for pre-existing leaks resulting in LERF magnitude releases is greatly reduced. For this assessment, for calculation of the Class 3a and 3b frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. A simplified separate effects containment spray model with a screening Human Error Probability of 0.1 for initiation of the containment sprays (as would be directed by SAG-5, Reduce Fission Product Releases, Rev. 0 [25]) was appended to the existing Level 1 sequence cutsets. This exercise revealed that 50% or more of the EPRI Class 1 CDF could result in sprays being available (even with accounting for dependent operator action failures and all hardware dependencies) thereby reducing the potential for LERF. This potential benefit was conservatively not credited in this analysis.

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

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC [6]) because it does not factor in the possibility that the

failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

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

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [19]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. Byron has a similar type of containment.

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

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

<u>Assumptions</u>

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 4.4-1, Step 1.)
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to this Byron containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2 were initiated from the non-visible (backside) portion of the containment liner.

- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis), and there is no evidence that additional corrosion issues were identified. (See Table 4.4-1, Step 1.)
- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages. (See Table 4.4-1, Steps 2 and 3.) Sensitivity studies are included that address doubling this rate every ten years and every two years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure. For Byron, the containment failure probabilities are much less than these values at the target pressures of 47.8 psig for Unit 1 and 44.4 psig for Unit 2. Conservative probabilities of 1% for the cylinder and dome and 0.1% for the basemat are used in this analysis, and sensitivity studies are included that increase and decrease the probabilities by an order of magnitude. (See Table 4.4-1, Step 4.)
- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the containment cylinder and dome region. (See Table 4.4-1, Step 4.)
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood

of 10% is used. To date, all liner corrosion events have been detected through visual inspection. (See Table 4.4-1, Step 5.) Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.

- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Table 4.4-1

Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome		Containment Basemat	
1	**Historical Steel Liner Flaw Likelihood**	Events: 2		Events: 0 (assume half a failure)	
	Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).	2/(70 * 5.5) = **5.2E-3**		0.5/(70 * 5.5) = **1.3E-3**	
2	Age Adjusted Steel Liner Flaw Likelihood During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).	Year	Failure Rate	Year	Failure Rate
		1	2.05E-03	1	5.13E-04
		avg 5-10	5.20E-03	avg 5-10	1.30E-03
		15	1.43E-02	15	3.51E-03
		15 year average = 6.14E-03		**15 year average = 1.54E-03**	

Table 4.4-1

Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome	Containment Basemat
3	Flaw Likelihood at 3, 10, and 15 years Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference [19]).	**0.71% (1 to 3 years)** **4.06% (1 to 10 years)** **9.40% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the desired presentation of the results.)	**0.18% (1 to 3 years)** **1.03% (1 to 10 years)** **2.39% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with desired presentation of the results.)
4	Likelihood of Breach in Containment Given Steel Liner Flaw The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.

Table 4.4-1

Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome	Containment Basemat
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0094% (at 15 years)** 9.4% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 100%

Analysis

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

Total Likelihood Of Non-Detected Containment Leakage Due To Corrosion:

- At 3 years: 0.00071% + 0.00018% = 0.00089%
- At 10 years: 0.0041% + 0.0010% = 0.0051%
- At 15 years: 0.0094% + 0.0024% = 0.0118%

Byron Past ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 under option B criteria is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3.

Based on completion of two successful ILRTs at each of the Byron units, the current ILRT interval is once per ten years. The next Type A test for Byron Unit 1 is currently due to be completed by 2/2008, and by 11/2009 for Unit 2.

Note that the probability of a pre-existing leakage due to extending the ILRT interval is based on the industry wide historical results as discussed in the NEI Guidance document, and the only portion of Byron specific information utilized is the fact that the current ILRT interval is once per ten years.

NEI Interim Guidance

This analysis uses the approach outlined in the NEI Interim Guidance. [3, 21] The nine steps of the methodology are:

1. Quantify the baseline (nominal three year ILRT interval) frequency per reactor year for the EPRI accident categories of interest. Note that EPRI categories 4, 5, and 6 are not affected by changes in ILRT test frequency.

2. Determine the containment leakage rates for EPRI categories 1 and 3 where category 3 is subdivided into categories 3a and 3b for "small" and "large" isolation failures, respectively.

3. Develop the baseline population dose (person-rem) for the applicable EPRI categories.

4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in Step (3) by the associated frequency calculated in Step (1).

5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.

6. Determine the population dose rate for the new surveillance intervals of interest.

7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.

8. Evaluate the risk impact in terms of LERF.

9. Evaluate the change in conditional containment failure probability.

The first seven steps of the methodology calculate the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The ninth and final step of the interim methodology calculates the change in containment failure probability, referred to the conditional containment failure probability, CCFP. The NRC has previously accepted similar calculations [7] for the change in CCFP as the basis for showing that the proposed change is consistent with the defense in depth philosophy. As such this last step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174.

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "small" leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).

5.0 RESULTS

The application of the approach based on NEI Interim Guidance [3, 21], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6, 7, 20, 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5-1 lists these accident classes.

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

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5-1

ACCIDENT CLASSES

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

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

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5-1.

Step 2 - Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

Step 3 - Evaluate risk impact of extending Type A test interval from 3 to 15 and 10 to 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

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

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

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

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

The frequencies for the severe accident classes defined in Table 5.1-1 were developed for Byron by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 4.2-1, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4. The results of applying this process are discussed after Table 5.1-1.

Table 5.1-1
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (BYRON BASE CASE)

EPRI/NEI Class	Byron Release Category	Unit 1 Frequency Based on Categorized Results (per yr)	Unit 2 Frequency Based on Categorized Results (per yr)
8	A	3.16E-06	3.12E-06
2	B_1	4.22E-08	3.92E-08
8	B_2	7.00E-08	7.01E-08
7	B_3	9.84E-08	2.02E-07
2	C_1	1.62E-10	1.62E-10
8	C_2	2.01E-08	2.01E-08
7	C_3	0	0
2	D	5.29E-07	6.03E-07
8	E	5.42E-09	2.11E-10
2	F_1	6.72E-09	4.62E-09
7	F_2	7.79E-07	1.55E-06
2	G	6.06E-09	6.06E-09
Total LERF		4.72E-06	5.62E-06
Total non-LERF		5.31E-05	5.17E-05
Total CDF		5.78E-05	5.73E-05

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially assigned from the Level 2 Release Categories not listed in Table 5.1-1, minus the EPRI/NEI Class 3a and 3b frequency, calculated below. For simplicity, and since Byron only tracks LERF versus non-LERF release categories, all non-LERF endstates are assumed to be in this bin. This may tend to under-predict the calculated plant-specific total dose and CCFP values, but will not change the calculated changes in population dose, LERF, and CCFP that are the determined in this risk impact assessment.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Byron Release Categories B_1, C_1, D, F_1, and G, listed in Table 5.1-1.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small ($2L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are determined as follows:

$PROB_{Class_3a}$ = probability of small pre-existing containment liner leakage
= 0.027 [see Section 4.3]
$PROB_{Class_3b}$ = probability of large pre-existing containment liner leakage
= 0.0027 [see Section 4.3]

As described in section 4.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions), or that would include containment spray operation such that a Large Release would be unlikely (currently not credited in this assessment).

For Unit 1:

Class_3a = 0.027 ∗ (CDF-Class 2-Class 8)
= 0.027 * (5.78E-05 – 5.85E-07 – 3.26E-06) = 1.46E-6/yr

Class_3b = 0.0027 ∗ (CDF-Class 2-Class 8)
= 0.0027 * (5.78E-05 – 5.85E-07 – 3.26E-06) = 1.46E-7/yr

For Unit 2:

Class_3a = 0.027 ∗ (CDF-Class 2-Class 8)
= 0.027 * (5.73E-05 – 6.53E-07 – 3.21E-06) = 1.44E-6/yr

Class_3b = 0.0027 ∗ (CDF-Class 2-Class 8)
= 0.0027 * (5.73E-05 – 6.53E-07 – 3.21E-06) = 1.44E-7/yr

For this analysis, the associated containment leakage for Class 3A is $10L_a$ and for Class 3B is $35L_a$. These assignments are consistent with the NEI Interim Guidance.

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the NEI Interim Guidance, however, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined from Release Categories B_3, C_3 and F_2 from the Byron Level 2 results.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency is determined from Release Categories A, B_2, C_2 and E from the Byron Level 2 results.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definitions of accident classes defined in EPRI-TR-104285 and the NEI Interim Guidance. Table 5.1-2 summarizes these accident frequencies by accident class for Byron.

Table 5.1-2
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (BYRON BASE CASE)

Accident Classes (Containment Release Type)	Accident Progression Bin (APB)	Description	Unit 1 Frequency (per Rx-yr)		Unit 2 Frequency (per Rx-yr)	
			NEI Methodology	NEI Methodology Plus Corrosion	NEI Methodology	NEI Methodology Plus Corrosion
1	6 & 7	No Containment Failure [(1)]	5.15E-05	5.15E-05	5.01E-05	5.01E-05
2	2	Large Isolation Failures (Failure to Close)	5.85E-07	5.85E-07	6.53E-07	6.53E-07
3a	10 La	Small Isolation Failures (liner breach)	1.46E-06	1.46E-06	1.44E-06	1.44E-06
3b	35 La	Large Isolation Failures (liner breach)	1.46E-07	1.46E-07	1.44E-07	1.45E-07
4	NA	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA	NA
5	NA	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA	NA
6	NA	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA	NA
7	4	Failures Induced by Phenomena (Early and Late)	8.77E-07	8.77E-07	1.76E-06	1.76E-06
8	5	Bypass (SGTR and ISLOCA)	3.26E-06	3.26E-06	3.21E-06	3.21E-06
CDF		All CET end states	5.78E-05	5.78E-05	5.73E-05	5.73E-05

(1) For simplicity, includes all non-LERF endstates for Byron. This may tend to under-predict the calculated plant-specific total dose and CCFP values, but will not change the calculated changes in population dose, LERF, and CCFP that are the determined in this risk impact assessment.

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

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences compared to the reference plant as described in Section 4.2, and summarized in Table 4.2-4. The results of applying these releases to the EPRI/NEI containment failure classification are as follows:

Class 1	=	5.18E+02 person-rem (at $1.0L_a$) = 5.18E+02 person-rem [(1)]
Class 2	=	6.69E+05[(2)]
Class 3a	=	5.18E+02 person-rem x $10L_a$ = 5.18E+03 person-rem [(3)]
Class 3b	=	5.18E+02 person-rem x $35L_a$ = 1.81E+04 person-rem [(3)]
Class 4	=	Not analyzed
Class 5	=	Not analyzed
Class 6	=	Not analyzed
Class 7	=	5.13E+05 person-rem [(4)]
Class 8	=	8.41E+05 person-rem [(5)]

(1) The derivation is described in Section 4.2 for Byron. Class 1 is assigned the dose from the "no containment failure" APBs from NUREG/CR-4551 (i.e., APB #6 and APB #7). The dose is calculated as an average of the dose for these bins from Table 4.2-2.

(2) The Class 2, containment isolation failures, dose is assigned from APB #2 (Early CF) from Table 4.2-2.

(3) The Class 3a and 3b dose are related to the leakage rate as shown. This is consistent with the NEI Interim Guidance.

(4) The Class 7 dose is assigned from APB #4 (Late CF) from Table 4.2-2.

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from APB #5 (Bypass) from Table 4.2-2.

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [2] containment failure classifications, and consistent with the NEI guidance [3] are provided in Table 5.2-1.

Table 5.2-1

BYRON POPULATION DOSE ESTIMATES

FOR POPULATION WITHIN 50 MILES

Accident Classes (Containment Release Type)	Accident Progression Bin (APB)	Description	Person-Rem (50 miles)
1	6 & 7	No Containment Failure	5.18E+02
2	2	Large Isolation Failures (Failure to Close)	6.69E+05
3a	$10L_a$	Small Isolation Failures (liner breach)	5.18E+03
3b	$35L_a$	Large Isolation Failures (liner breach)	1.81E+04
4	N/A	Small Isolation Failures (Failure to seal –Type B)	NA
5	N/A	Small Isolation Failures (Failure to seal—Type C)	NA
6	N/A	Other Isolation Failures (e.g., dependent failures)	NA
7	4	Failures Induced by Phenomena (Early and Late)	5.13E+05
8	5	Bypass (SGTR and Interfacing System LOCA)	8.41E+05

The above dose estimates, when combined with the results presented in Table 5.1-2, yield the Byron baseline mean consequence measures for each accident class. These results are presented in Table 5.2-2 for Unit 1 and Table 5.2-3 for Unit 2.

Table 5.2-2

BYRON UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	5.18E+02	5.15E-05	2.66E-02	5.15E-05	2.66E-02	-2.48E-07
2	Large Isolation Failures (Failure to Close)	6.69E+05	5.85E-07	3.91E-01	5.85E-07	3.91E-01	--
3a	Small Isolation Failures (liner breach)	5.18E+03	1.46E-06	7.54E-03	1.46E-06	7.54E-03	--
3b	Large Isolation Failures (liner breach)	1.81E+04	1.46E-07	2.64E-03	1.46E-07	2.65E-03	8.68E-06
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	5.13E+05	8.77E-07	4.50E-01	8.77E-07	4.50E-01	--
8	Bypass (SGTR and ISLOCA)	8.41E+05	3.26E-06	2.74	3.26E-06	2.74	--
CDF	All CET end states		5.78E-05	3.62	5.78E-05	3.62	8.43E-06

1) Only release Classes 1 and 3b are affected by the corrosion analysis.

2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.2-3

BYRON UNIT 2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	5.18E+02	5.01E-05	2.59E-02	5.01E-05	2.59E-02	-2.45E-07
2	Large Isolation Failures (Failure to Close)	6.69E+05	6.53E-07	4.37E-01	6.53E-07	4.37E-01	--
3a	Small Isolation Failures (liner breach)	5.18E+03	1.44E-06	7.46E-03	1.44E-06	7.46E-03	--
3b	Large Isolation Failures (liner breach)	1.81E+04	1.44E-07	2.61E-03	1.45E-07	2.62E-03	8.59E-06
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	5.13E+05	1.76E-06	9.01E-01	1.76E-06	9.01E-01	--
8	Bypass (SGTR and ISLOCA)	8.41E+05	3.21E-06	2.70	3.21E-06	2.70	--
CDF	All CET end states		5.73E-05	4.07	5.73E-05	4.07	8.34E-06

1) Only release Classes 1 and 3b are affected by the corrosion analysis.

2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

As indicated previously, the total dose may be slightly under-predicted due to the treatment of all non-LERF endstates as being assigned to EPRI Class 1. Although the total dose may be under-predicted, the Byron dose would still compare favorably with other locations given the relative population densities surrounding each location:

Plant	Annual Dose (Person-Rem/Yr)	Reference
Indian Point 3	14,515	[6]
Peach Bottom	6.2	[23]
Crystal River	1.4	[20]
Byron Unit 1	3.6	[Table 5.2-2]
Byron Unit 2	4.1	[Table 5.2-3]

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

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to fifteen-years. To do this, an evaluation must first be made of the risk associated with the ten-year interval since the base case applies to a 3-year interval (i.e., a simplified representation of a 3-in-10 interval).

Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the NEI guidance as described in Section 4.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 5.3-1 for Unit 1 and Table 5.3-2 for Unit 2.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value, as described in Section 4.3. The results for this calculation are presented in Table 5.3-3 for Unit 1 and Table 5.3-4 for Unit 2.

Table 5.3-1

BYRON UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	5.18E+02	4.78E-05	2.47E-02	4.78E-05	2.47E-02	-1.42E-06
2	Large Isolation Failures (Failure to Close)	6.69E+05	5.85E-07	3.91E-01	5.85E-07	3.91E-01	--
3a	Small Isolation Failures (liner breach)	5.18E+03	4.85E-06	2.51E-02	4.85E-06	2.51E-02	--
3b	Large Isolation Failures (liner breach)	1.81E+04	4.85E-07	8.79E-03	4.88E-07	8.84E-03	4.96E-05
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	5.13E+05	8.77E-07	4.50E-01	8.77E-07	4.50E-01	--
8	Bypass (SGTR and ISLOCA)	8.41E+05	3.26E-06	2.74	3.26E-06	2.74	--
CDF	All CET end states		5.78E-05	3.64	5.78E-05	3.64	4.82E-05

[1] Only release classes 1 and 3b are affected by the corrosion analysis.

[2] Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.3-2

BYRON UNIT 2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	5.18E+02	4.64E-05	2.40E-02	4.64E-05	2.40E-02	-1.40E-06
2	Large Isolation Failures (Failure to Close)	6.69E+05	6.53E-07	4.37E-01	6.53E-07	4.37E-01	--
3a	Small Isolation Failures (liner breach)	5.18E+03	4.80E-06	2.49E-02	4.80E-06	2.49E-02	--
3b	Large Isolation Failures (liner breach)	1.81E+04	4.80E-07	8.70E-03	4.83E-07	8.75E-03	4.91E-05
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	5.13E+05	1.76E-06	9.01E-01	1.76E-06	9.01E-01	--
8	Bypass (SGTR and ISLOCA)	8.41E+05	3.21E-06	2.70	3.21E-06	2.70	--
CDF	All CET end states		5.73E-05	4.09	5.73E-05	4.09	4.77E-05

[1] Only release classes 1 and 3b are affected by the corrosion analysis.

[2] Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.3-3

BYRON UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	5.18E+02	4.51E-05	2.33E-02	4.51E-05	2.33E-02	-3.28E-06
2	Large Isolation Failures (Failure to Close)	6.69E+05	5.85E-07	3.91E-01	5.85E-07	3.91E-01	--
3a	Small Isolation Failures (liner breach)	5.18E+03	7.29E-06	3.77E-02	7.29E-06	3.77E-02	--
3b	Large Isolation Failures (liner breach)	1.81E+04	7.29E-07	1.32E-02	7.35E-07	1.33E-02	1.15E-04
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	5.13E+05	8.77E-07	4.50E-01	8.77E-07	4.50E-01	--
8	Bypass (SGTR and ISLOCA)	8.41E+05	3.26E-06	2.74	3.26E-06	2.74	--
CDF	All CET end states		5.78E-05	3.65	5.78E-05	3.65	1.12E-04

(1) Only release classes 1 and 3b are affected by the corrosion analysis.

(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5.3-4

BYRON UNIT 2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr[1]
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure [2]	5.18E+02	4.37E-05	2.26E-02	4.37E-05	2.26E-02	-3.25E-06
2	Large Isolation Failures (Failure to Close)	6.69E+05	6.53E-07	4.37E-01	6.53E-07	4.37E-01	--
3a	Small Isolation Failures (liner breach)	5.18E+03	7.21E-06	3.73E-02	7.21E-06	3.73E-02	--
3b	Large Isolation Failures (liner breach)	1.81E+04	7.21E-07	1.31E-02	7.27E-07	1.32E-02	1.14E-04
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal— Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	5.13E+05	1.76E-06	9.01E-01	1.76E-06	9.01E-01	--
8	Bypass (SGTR and ISLOCA)	8.41E+05	3.21E-06	2.70	3.21E-06	2.70	--
CDF	All CET end states		5.73E-05	4.11	5.73E-05	4.11	1.10E-04

(1) Only release classes 1 and 3b are affected by the corrosion analysis.

(2) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.4 Step 4 – Determine the Change in Risk in Terms of Large Early Release Frequency

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the NEI guidance, 100% of the Class 3b contribution would be considered LERF. For Byron, however, the Class 3b radionuclide release person-rem is significantly less than a typical LERF contributor as can be seen by comparing the relative population dose for Class 3b to that of Class 2 (1.81E+04 person-rem / 6.69E+05 person-rem or 2.7%). Additionally, as was noted in Section 4.3, a substantial portion of this increase could potentially be non-LERF contributors if the availability of containment sprays were factored into the analysis. As such, based on these two considerations, it should be recognized that classifying all of the Class 3b contributions as LERF is very conservative.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Byron, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the NEI guidance methodology). Based on the original 3/10 year test interval assessment from Tables 5.2-2 and 5.2-3, the Class 3b frequency is 1.46E-07/yr for Unit 1 and 1.45E-07 for Unit 2. Based on a ten-year test interval from Tables 5.3-1 and 5.3-2, the Class 3b frequency is 4.88E-07/yr for Unit 1 and 4.83E-07/yr for Unit 2; and, based on a fifteen-year test interval from Tables 5.3-3 and 5.3-4, it is 7.35E-07/yr for Unit 1 and 7.27E-07/yr for Unit 2.

Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 5.89E-07/yr for Unit 1 and 5.82E-07/yr for Unit 2. Similarly, the increase due to increasing the interval from 10 to 15 years is 2.47E-07/yr for Unit 1 and 2.44E-07/yr for Unit 2. As can be seen, even with the conservatisms included in the evaluation (per the NEI methodology), the estimated change in LERF is below the threshold criteria for a small change in risk when comparing the 15 year results to the current 10-year requirement or to the original 3-in-10 year requirement.

5.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability

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

The change in CCFP can be calculated by using the method specified in the NEI Interim Guidance. The NRC has previously accepted similar calculations [7] as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy.

Unit	CCFP 3 in 10 yrs	CCFP 1 in 10 yrs	CCFP 1 in 15 yrs	$\Delta CCFP_{15\text{-}3}$	$\Delta CCFP_{15\text{-}10}$
1	8.41%	9.00%	9.42%	1.01%	0.42%
2	10.06%	10.65%	11.07%	1.01%	0.42%

CCFP = [1 - (Class 1 frequency + Class 3a frequency) / CDF] * 100%

The change in CCFP of slightly more than 1% by extending the test interval to 15 years from the original 3-in-10 year requirement is judged to be insignificant.

5.6 Summary of Results

The results from this ILRT extension risk assessment for Byron are summarized in Table 5.6-1 for Unit 1 and Table 5.6-2 for Unit 2.

Table 5.6-1

Byron Unit 1 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	5.18E+02	5.15E-05	2.66E-02	4.78E-05	2.47E-02	4.51E-05	2.33E-02
2	6.69E+05	5.85E-07	3.91E-01	5.85E-07	3.91E-01	5.85E-07	3.91E-01
3a	5.18E+03	1.46E-06	7.54E-03	4.85E-06	2.51E-02	7.29E-06	3.77E-02
3b	1.81E+04	1.46E-07	2.65E-03	4.88E-07	8.84E-03	7.35E-07	1.33E-02
7	5.13E+05	8.77E-07	4.50E-01	8.77E-07	4.50E-01	8.77E-07	4.50E-01
8	8.41E+05	3.26E-06	2.74	3.26E-06	2.74	3.26E-06	2.74
Total		5.78E-05	3.62	5.78E-05	3.64	5.78E-05	3.65
ILRT Dose Rate from 3a and 3b		1.02E-02		3.40E-02		5.10E-02	
Delta Total Dose Rate	From 3 yr	---		2.38E-02		4.08E-02	
	From 10 yr	---		---		1.71E-02	
3b Frequency (LERF)		1.46E-07		4.88E-07		7.35E-07	
Delta LERF	From 3 yr	---		3.42E-07		5.89E-07	
	From 10 yr	---		---		2.47E-07	
CCFP %		8.41%		9.00%		9.42%	
Delta CCFP %	From 3 yr	---		0.60%		1.01%	
	From 10 yr	---		---		0.42%	

Table 5.6-2

Byron Unit 2 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	5.18E+02	5.01E-05	2.59E-02	4.64E-05	2.40E-02	4.37E-05	2.26E-02
2	6.69E+05	6.53E-07	4.37E-01	6.53E-07	4.37E-01	6.53E-07	4.37E-01
3a	5.18E+03	1.44E-06	7.46E-03	4.80E-06	2.49E-02	7.21E-06	3.73E-02
3b	1.81E+04	1.45E-07	2.62E-03	4.83E-07	8.75E-03	7.27E-07	1.32E-02
7	5.13E+05	1.76E-06	9.01E-01	1.76E-06	9.01E-01	1.76E-06	9.01E-01
8	8.41E+05	3.21E-06	2.70	3.21E-06	2.70	3.21E-06	2.70
Total		5.73E-05	4.07	5.73E-05	4.09	5.73E-05	4.11
ILRT Dose Rate from 3a and 3b		1.01E-02		3.36E-02		5.05E-02	
Delta Total Dose Rate	From 3 yr	---		2.35E-02		4.04E-02	
	From 10 yr	---		---		1.69E-02	
3b Frequency (LERF)		1.45E-07		4.83E-07		7.27E-07	
Delta LERF	From 3 yr	---		3.38E-07		5.83E-07	
	From 10 yr	---		---		2.44E-07	
CCFP %		10.06%		10.65%		11.07%	
Delta CCFP %	From 3 yr	---		0.59%		1.01%	
	From 10 yr	---		---		0.42%	

6.0 SENSITIVITIES

6.1 Sensitivity to Corrosion Impact Assumptions

The results in Tables 5.6-1, 5.6-2 and 6.1-1 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the cylinder and dome and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6.1-1. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 1.88E-7/yr for Unit 1 and 1.86E-7/yr for Unit 2. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 6.1-1
Steel Liner Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)	Unit 1 Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 years (per Rx-yr)		Unit 2 Increase in Class 3b Frequency (LERF) for ILRT Extension 3 to 15 years (per Rx-yr)	
			Total Increase	Increase Due to Corrosion	Total Increase	Increase Due to Corrosion
Base Case Doubles every 5 yrs	Base Case (1% Cylinder, 0.1% Basemat)	Base Case 10%	5.89E-07	5.86E-09	5.83E-07	5.80E-09
Doubles every 2 yrs	Base	Base	5.96E-07	1.34E-08	5.90E-07	1.33E-08
Doubles every 10 yrs	Base	Base	5.88E-07	4.94E-09	5.82E-07	4.89E-09
Base	Base	15%	5.91E-07	8.21E-09	5.85E-07	8.12E-09
Base	Base	5%	5.86E-07	3.52E-09	5.80E-07	3.48E-09
Base	10% Cylinder, 1% Basemat	Base	6.42E-07	5.86E-08	6.35E-07	5.80E-08
Base	0.1% Cylinder, 0.01% Basemat	Base	5.84E-07	5.86E-10	5.780E-07	5.80E-10
Lower Bound						
Doubles every 10 yrs	0.1% Cylinder, 0.01% Basemat	5% 100%	5.83E-07	2.97E-10	5.77E-07	2.94E-10
Upper Bound						
Doubles every 2 yrs	10% Cylinder, 1% Basemat	15% 100%	7.71E-07	1.88E-07	7.63E-07	1.86E-07

6.2 EPRI Expert Elicitation Sensitivity

An expert elicitation was performed to reduce excess conservatisms in the data associated with the probability of undetected leak within containment [22]. Since the risk impact assessment of the extensions to the ILRT interval is sensitive to both the probability of the leakage as well as the magnitude, it was decided to perform the expert elicitation in a manner to solicit the probability of leakage as function of leakage magnitude. In addition, the elicitation was performed for a range of failure modes which allowed experts to account for the range of mechanisms of failure, the potential for undiscovered mechanisms, un-inspectable areas of the containment as well as the potential for detection by alternate means. 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 basic difference in the application of the ILRT interval methodology using the expert elicitation is a change in the probability of pre-existing leakage in the containment. The basic methodology uses the Jeffery's 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 L_a for small and 35 L_a for large) are used here. Table 6.2-1 illustrates the magnitudes and probabilities of a pre-existing leak in containment associated with the Jeffery's non-informative prior and the expert elicitation statistical treatments. These values are used in the ILRT interval extension for the base methodology and in this sensitivity case. Details of the expert elicitation process, the input to expert elicitation as well as the results of the expert elicitation are available in the various appendices of the EPRI report [22].

Table 6.2-1
EPRI Expert Elicitation Results

Leakage Size (La)	Jeffery's Non-Informative Prior	Expert Elicitation Mean Probability of Occurrence	Percent Reduction
10	2.7E-02	3.88E-03	86%
35	2.7E-03	9.86E-04	64%

A summary of the results using the expert elicitation values for probability of containment leakage is provided in Table 6.2-2 for Unit 1 and Table 6.2-3 for Unit 2. As mentioned previously, probability values are those associated with the magnitude of the leakage used in the Jeffery's non-informative prior evaluation ($10L_a$ for small and $35L_a$ for large). The expert elicitation process produces a probability versus leakage magnitude relationship and it is possible to assess higher leakage magnitudes more reflective of large early releases but these evaluations are not performed in this study. Alternative leakage magnitudes could include consideration of 100 – to 600 L_a where leakage begins to approach large early releases.

The net affect is that the reduction in the multipliers shown above has the same impact on the calculated increases in the LERF values. The increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 2.13E-07/yr for Unit 1 and 2.11E-07/yr for Unit 2. Similarly, the increase due to increasing the interval from 10 to 15 years is 8.89E-08/yr for Unit 1 and 8.80E-08/yr for Unit 2. As such, if the expert elicitation mean probability of occurrences are used instead of the non-informative prior estimates, the change in LERF for Byron is below the threshold criteria for a "very small" change in risk when comparing the 15 year results to the current 10-year requirement and is just above the "very small" change threshold value of 1.0E-7/yr in the "small" change region when compared to the original 3-in-10 year requirement. The results of this sensitivity study are judged

to be more indicative of the actual risk associated with the ILRT extension than the results from the assessment as dictated by the NEI methodology values, and yet are still conservative given the assumption that all of the Class 3b contribution is considered to be LERF.

Table 6.2-2

Byron Unit 1 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions (Based on EPRI Expert Elicitation Leakage Probabilities)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	5.18E+02	5.28E-05	2.73E-02	5.22E-05	2.70E-02	5.18E-05	2.68E-02
2	6.69E+05	5.85E-07	3.91E-01	5.85E-07	3.91E-01	5.85E-07	3.91E-01
3a	5.18E+03	2.09E-07	1.08E-03	6.97E-07	3.61E-03	1.05E-06	5.42E-03
3b	1.81E+04	5.32E-08	9.64E-04	1.77E-07	3.21E-03	2.66E-07	4.82E-03
7	5.13E+05	8.77E-07	4.50E-01	8.77E-07	4.50E-01	8.77E-07	4.50E-01
8	8.41E+05	3.26E-06	2.74	3.26E-06	2.74	3.26E-06	2.74
Total		5.78E-05	3.61	5.78E-05	3.61	5.78E-05	3.62
ILRT Dose Rate from 3a and 3b		2.05E-03		6.82E-03		1.02E-02	
Delta Total Dose Rate	From 3 yr	---		4.45E-03		7.65E-03	
	From 10 yr	---		---		3.19E-03	
3b Frequency (LERF)		5.32E-08		1.77E-07		2.66E-07	
Delta LERF	From 3 yr	---		1.24E-07		2.13E-07	
	From 10 yr	---		---		8.89E-08	
CCFP %		8.25%		8.47%		8.62%	
Delta CCFP %	From 3 yr	---		0.21%		0.37%	
	From 10 yr	---		---		0.15%	

Table 6.2-3

Byron Unit 2 ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions ((Based on EPRI Expert Elicitation Leakage Probabilities)

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	5.18E+02	5.14E-05	2.66E-02	5.08E-05	2.63E-02	5.04E-05	2.61E-02
2	6.69E+05	6.53E-07	4.37E-01	6.53E-07	4.37E-01	6.53E-07	4.37E-01
3a	5.18E+03	2.07E-07	1.07E-03	6.90E-07	3.57E-03	1.04E-06	5.36E-03
3b	1.81E+04	5.27E-08	9.54E-04	1.75E-07	3.18E-03	2.63E-07	4.77E-03
7	5.13E+05	1.76E-06	9.01E-01	1.76E-06	9.01E-01	1.76E-06	9.01E-01
8	8.41E+05	3.21E-06	2.70	3.21E-06	2.70	3.21E-06	2.70
Total		5.73E-05	4.07	5.73E-05	4.07	5.73E-05	4.07
ILRT Dose Rate from 3a and 3b		2.03E-03		6.75E-03		1.01E-02	
Delta Total Dose Rate	From 3 yr	---		4.41E-03		7.57E-03	
	From 10 yr	---		---		3.16E-03	
3b Frequency (LERF)		5.27E-08		1.75E-07		2.63E-07	
Delta LERF	From 3 yr	---		1.23E-07		2.11E-07	
	From 10 yr	---		---		8.80E-08	
CCFP %		9.90%		10.11%		10.27%	
Delta CCFP %	From 3 yr	---		0.21%		0.37%	
	From 10 yr	---		---		0.15%	

6.3 Potential Impact from External Events Contribution

In the Byron IPEEE, the dominant risk contributor from external events was found to be from fire events. Other potential contributors such as seismic and high winds were found to be within acceptable limits. As a reasonable assessment of the impact from external events, one can assume that the external events CDF is comparable to the internal events CDF. Additionally, one can assume that the fractional LERF contribution from the internal events model (excluding the contribution from ISLOCA or SGTR scenarios since these types of events would typically not occur from an external event initiator) also provides a reasonable estimate of the LERF impact from external events.

For Byron Unit 1, the reported total Internal Events LERF as determined from a simplified LERF model is 4.72E-06/yr, and for Unit 2 it is 5.62E-06/yr [16]. As indicated above, the External Events baseline LERF would be expected to be less than the Internal Events baseline LERF because some of the Internal Events baseline LERF comes from events that are not events that are initiated by fires (i.e., ISLOCA and SGTR). Subtracting off the contributions from these events (i.e. EPRI Class 8) of 3.26E-6/yr for Unit 1 and 3.21E-6/yr for Unit 2 yields a LERF value subject to the External Events impact of 1.46E-6/yr for Unit 1 and 2.41E-6/yr for Unit 2. There are some known conservatisms in the simplified LERF model, but these values will be used in the discussion below for illustration purposes.

However, as is shown in Table 6.3-1, if it is assumed that the LERF impact of the ILRT extension from External Events is assumed to be the same as that from Internal Events, the total LERF would be below the Regulatory Guide 1.174 criteria of 1.0E-05 following the ILRT extension. Using the same assumptions, Table 6.3-2 shows the impact if the EPRI expert elicitation values are used for the Class 3a and 3b frequency determination. In this case, the total LERF is further below the Regulatory Guide 1.174 criteria of 1.0E-05 following the ILRT extension.

Table 6.3-1
Byron Estimated Total LERF Including External Events Impact
(Base Case NEI Methodology)

Contributor	Byron Unit 1	Byron Unit 2
Internal Events LERF	4.72E-06	5.62E-06
External Events LERF	1.46E-06	2.41E-06
Internal Events LERF due to ILRT (at 15 years)	7.35E-07	7.27E-07
External Events LERF due to ILRT (at 15 years)	7.35E-07	7.27E-07
Total:	**7.65E-06**	**9.48E-06**

Table 6.3-2
Byron Estimated Total LERF Including External Events Impact
(EPRI Expert Elicitation Methodology)

Contributor	Byron Unit 1	Byron Unit 2
Internal Events LERF	4.72E-06	5.62E-06
External Events LERF	1.46E-06	2.41E-06
Internal Events LERF due to ILRT (at 15 years)	2.66E-07	2.63E-07
External Events LERF due to ILRT (at 15 years)	2.66E-07	2.63E-07
Total:	**6.71E-06**	**8.56E-06**

7.0 CONCLUSIONS

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

- Reg. Guide 1.174 [4] 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 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 internal events LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is very conservatively estimated as 5.9E-7/yr for Unit 1 and 5.8E-7/yr for Unit 2 using the NEI guidance as written, and at 2.1E-7/yr for both Units using the EPRI Expert Elicitation methodology. These values could also be reduced if the potential impact from the availability of containment sprays were factored into the analysis, but even without accounting for this reduction, the estimated change in LERF is determined to be "small" using the acceptance guidelines of Reg. Guide 1.174.
- Regulatory Guide 1.174 [4] also states that when the calculated increase in LERF is in the range of 1.0E-06 per reactor year to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. As such, an additional assessment of the impact from external events was also made. In that case, the total LERF was conservatively estimated as 7.7E-06/yr and 9.5E-06/yr for Byron Units 1 and 2, respectively using the NEI guidance directly. These numbers fall to 6.7E-6/yr and 8.6E-6/yr if the EPRI Expert Elicitation methodology is utilized. These values are all below the RG 1.174 acceptance criteria for total LERF of 1.0E-05, but the EPRI Expert Elicitation

methodology provides more margin to the limit than that provided by the NEI methodology directly.

- The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.04 person-rem/yr for both Units using the NEI guidance, and drops to 0.008 person-rem/yr using the EPRI Expert Elicitation methodology. Therefore, in either case, the risk impact when compared to other severe accident risks is negligible.
- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen year interval is about 1% using the NEI guidance, and drops to about 0.4% using the EPRI Expert Elicitation methodology. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the Byron Station risk profile.

Previous Assessments

The NRC in NUREG-1493 [5] has previously concluded that:

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

containment penetrations, ILRTs also test the integrity of the containment structure.

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

8.0 REFERENCES

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Appendix A CDF and LERF Subcategory Calculations

CDF is available from Rev. 5E Quantification Notebook [Ref. 1].

LERF Sequence frequencies are calculated using the following process:

1. Quantify each sequence using PRAQuant at a truncation limit of 1E-11. Results are stored in separate cutset files (i.e., one sequence per file). This is done so that each cutset is tagged with a "class" label that identifies the sequence. PRAQuant files are listed in Reference 6.
2. Use Merger32.exe [Ref. 4] to combine sequences into one cutset file. This needs to be done in order to subsume non-minimal or duplicate cutsets (i.e., cutsets that satisfy more than one sequence logic).
3. Use CSUTIL32.exe [Ref. 5] "Set Event Flags" feature on the merged cutset file. Set 1.0 events to TRUE and subsume. The result of this action is a cutset file that matches the base model quantification at the same truncation frequency.
4. Use CSUTIL32.exe "Split Classes" feature to split the cutsets into classes (that represent sequences). The results of this action are a cutset file containing all sequences with a separate module for each sequence, and a listing of each class (sequence) with the number of cutsets and total LERF for that sequence. This file is sent to the clipboard by CSUTIL32 and can be pasted into Excel.

After the LERF sequence frequencies are quantified, and the results stored in an Excel spreadsheet, the LERF category frequencies are obtained using the following process:

1. Note that each sequence designator (class) ends with a letter that represents the LERF category associated with that Level 1 sequence. The sequence designator, and hence the LERF category assignment is provided in Table D.3 of the Quantification Notebook (BB PRA-014), Appendix D [Ref. 1]
2. LERF Category B is divided into B_1, B_2 and B_3. The contribution of each sequence to subcategories B_2 and B_3 is determined using the Fussel-Vesely importance of the following basic events, multiplied by the total sequence frequency:

   B_2: ISGTR

   B_3: CF-VB1-U1 or CF-VB1-U2
3. The frequency of LERF Category B_1 is the remainder of the sequence frequency not assigned to B_2 and B_3.

4. LERF Category C is divided into C_1, C_2 and C_3. The contribution of each sequence to these subcategories is determined the same way as the Category B subcategories.
5. LERF Category F is divided into F_1 and F_2. The contribution of each sequence to subcategory F_2 is determined using the Fussel-Vesely importance for ISGTR multiplied by the total sequence frequency. Subcategory F_1 frequency is the remainder of the sequence frequency not assigned to F_2.
6. The sum totals for the category and subcategory frequencies from the Excel spreadsheet [Ref. 2] are copied to Table A-1. These values are also used in Table 4.2-1 where the frequency assignments to the different accident progression bins and EPRI release categories are made to perform the ILRT extension assessment.

Sensitivity for URE BB-591

URE BB-591 relates to missing power supplies for certain containment isolation valves in the Unit 2 model. If the model were corrected to accurately reflect the power supplies for these valves, the calculations performed for the ILRT extension would be affected. Therefore, a sensitivity analysis was performed to determine the impact of this model error on the ILRT calculations.

1. The Revision 5E fault tree (Master5E.caf) [Ref. 1] was modified to include the following dependencies:

Valve	Bus
2SI8814	MCC 231X1A
2SI8920	MCC 231X1A
2SI8813	MCC 232X4A

2. The model quantifications and manipulations described in the previous section were repeated for Unit 2.
3. The calculations performed by the spreadsheet, as described in the previous section, were performed for the new LERF results. This includes updating the Fussell-Vesely importance values for ISGTR and Containment Failure at Vessel Breach, for those sequences with frequency changes as a result of the sensitivity.
4. The results of the new LERF category calculations [Ref. 3] are shown in Table A-1. As can be seen, there are insignificant changes to a few of the LERF subcategories. Therefore, the results of the original calculations would not be significantly impacted and the conclusions remain valid.

Table A-1 Byron LERF Model Subcategory Development

Byron LERF Category	Definition	Unit 1 Frequency/ yr	Unit 2 Frequency /yr	Unit 2 Sensitivity Frequency	Unit 2 Sensitivity Delta
A	Straight pass through CDF sequence to LERF	3.16E-06	3.12E-06	3.12E-06	0
B_1	High pressure sequences with no AFW available.	4.22E-08	3.92E-08	4.08E-08	2E-09
B_2	High pressure sequences with no AFW available, where the possibility exists for an ISGTR.	7.00E-08	7.01E-08	7.02E-08	1E-10
B_3	High pressure sequences with no AFW available and questions whether the containment fails at the time of vessel breach.	9.84E-08	2.02E-07	2.03E-07	1E-09
C_1	High pressure sequences with no AFW available.	1.62E-10	1.62E-10	1.62E-10	0
C_2	High pressure sequences with no AFW available, where the possibility exists for an ISGTR.	2.01E-08	2.01E-08	2.01E-08	0
C_3	High pressure sequences with no AFW available and the time between core uncovery and vessel breach is greater than the required evacuation time.	0	0	0	0
D	Sequences that do not lead to containment failure or result in containment failure many hours after core uncovery. Containment isolation is asked.	5.29E-07	6.03E-07	6.05E-07	2E-09
E	SGTR sequences where isolation of the ruptured SG is possible but has not been questioned in the Level 1 event tree, and asks to see isolation of the ruptured SG is successful.	5.42E-09	2.11E-10	2.11E-10	0
F_1	High pressure sequences with AFW available.	6.72E-09	4.62E-09	4.62E-09	0
F_2	High pressure sequences with AFW available, where no possibility exists for an ISGTR.	7.79E-07	1.55E-06	1.55E-06	0
G	Sequences where the RCS pressure is low at the time of vessel breach and AFW available, where no possibility exists for an ISGTR.	6.06E-09	6.06E-09	6.06E-09	0

Appendix A References

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B11-5E-sc.qnt	46 KB	8/7/2006 4:05 pm
B11-5E-sl.qnt	46 KB	8/7/2006 8:39 pm
b22-5E-sCDF.qnt	46 KB	8/8/2006 10:58 am
b22-5E-sLERF.qnt	46 KB	8/8/2006 12:04 pm