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10 CFR 50.90

August 28, 2009

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Peach Bottom Atomic Power Station, Unit 2
Renewed Facility Operating License No. DPR-44
NRC Docket No. 50-277

Subject: License Amendment Request - Type A Test Extension

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC (EGC) hereby requests a proposed change to modify Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program." Specifically, the proposed change will revise TS 5.5.12 to reflect a one-time extension of the containment Type A Integrated Leak Rate Test (ILRT) from 10 to 15 years. This one-time extension will require the Type A ILRT to be performed no later than October 2015.

The proposed change has been reviewed by the Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed amendment by August 28, 2010. Once approved, the amendment shall be implemented within 60 days.

No commitments are contained in this request.

In accordance with 10 CFR 50.91, EGC is notifying the State of Pennsylvania of this application for changes to the TS and Operating Licenses by transmitting a copy of this letter and its attachments to the designated state official.

Should you have any questions concerning this letter, please contact Tom Loomis at (610) 765-5510.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 28th of August 2009.

Respectfully,

DOK

Pamela B. Cowan
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
(1) Evaluation of Proposed Change
(2) Markup of Proposed Technical Specification Page Change
(3) Retyped Page for Technical Specification Change
(4) Risk Assessment for Peach Bottom Unit 2 To Support ILRT (Type A) Interval Extension Request

cc:
S. J. Collins, Regional Administrator, Region I, USNRC
S. T. Gray, State of Maryland
F. Bower, USNRC Senior Resident Inspector, PBAPS
J. Hughey, Project Manager, USNRC
R. R. Janati, Commonwealth of Pennsylvania

Attachment 1

Peach Bottom Atomic Power Station, Unit 2

Facility Operating License No. DPR-44

Evaluation of Proposed Change

ATTACHMENT 1 CONTENTS

SUBJECT: Type A Test Extension

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Facility Operating License No. DPR-44 for Peach Bottom Atomic Power Station (PBAPS), Unit 2.

The proposed change modifies Technical Specification (TS) Section 5.5.12 to reflect a one-time extension of the Type A containment Integrated Leak Rate Test (ILRT) to no later than October 2015. Exelon Generation Company, LLC (EGC) requests approval of the proposed change by August 28, 2010. Once approved, the amendment shall be implemented within 60 days.

2.0 DETAILED DESCRIPTION

The proposed change involves a one-time extension to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 1). The current ten (10) year containment Integrated Leak Rate Test (ILRT) for PBAPS, Unit 2 is due in October 2010 and is currently scheduled to be performed during Refueling Outage 2R18 in 2010. The proposed exception would allow the next ILRT for PBAPS, Unit 2 to be performed within fifteen (15) years (October 2015) from the last ILRT as opposed to the current ten (10) year frequency.

The proposed change would revise Section 5.5.12 ("Primary Containment Leakage Rate Testing Program") of the PBAPS, Unit 2 Technical Specifications to add the following statement:

"b. Section 9.2.3: The first Type A test performed after the October 2000 Type A test shall be performed no later than October 2015."

2.1 Background

The proposed change involves a one-time extension to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 1). The most recent containment Type A ILRT for PBAPS, Unit 2 was performed in October 2000 and would need to be performed no later than Refueling Outage 2R18 in 2010. The proposed exception would allow the next Type A ILRT to be performed within fifteen (15) years (i.e., October 2015) as opposed to the current ten (10) year frequency.

This one-time extension will result in the following:

- Perform a Type A ILRT no later than October 2015.
- A substantial cost savings will be realized by deferring the Type A test for an additional five (5) years. Cost savings have been estimated at approximately $2.3 million, which includes labor, equipment and critical path outage time needed to perform the test.

2.2 Description of Primary Containment System

As discussed in Section 5.2.3.1 of the PBAPS, Units 2 and 3 Updated Final Safety Analysis Report (UFSAR), the Primary Containment is a pressure suppression system and houses the reactor vessel, the reactor coolant recirculation systems, and other primary system piping. The Primary Containment system consists of a Drywell, a pressure Suppression Chamber which stores a large volume of water, a connecting vent system between the Drywell and the suppression pool, isolation valves, vacuum breakers, containment cooling systems, and other service equipment. The Primary Containment is designed for a maximum internal pressure of 62 psig coincident with a maximum temperature of 281°F. The maximum external pressure is 2 psi above internal pressure.

Vacuum breakers are provided in the vent headers and located in the Suppression Chamber to equalize the pressure between the Drywell and the Suppression Chamber. A vacuum breaker system is also provided between the Suppression Chamber and Secondary Containment. Cooling systems are provided to remove heat from the Drywell and from the water in the Suppression Chamber. Appropriate isolation valves are provided to ensure containment of radioactive materials.

The vent system conducts flow from the Drywell to the Suppression Chamber and distributes this flow uniformly in the suppression pool. The suppression pool condenses the steam portion of this flow and the Suppression Chamber contains the non-condensable gases and fission products. The Suppression Chamber-to-Drywell vacuum breakers and the Suppression Chamber-to-Secondary Containment vacuum breaker system limit the pressure differential so as not to exceed the design limit of 2 psi. The Suppression Chamber is designed for the same leakage rate as the Drywell.

The Primary Containment was designed, fabricated, and inspected in compliance with the requirements of ASME Boiler and Pressure Vessel Code, Section III, Subsection B (1965) with all applicable Addenda through Summer 1966.

As discussed in UFSAR Section 5.2.3.2 ("Drywell"), the Drywell is a light bulb-shaped steel pressure vessel with a spherical lower portion, 67 ft in diameter, and a cylindrical upper portion 38 ft 6 inches in diameter. The overall height is approximately 114 ft. The Drywell is enclosed in reinforced concrete for shielding purposes. Above the Drywell foundation, the concrete is separated from the containment vessel by an air gap of approximately 2 inches. As also discussed in UFSAR Section 5.2.3.3 ("Pressure Suppression Chamber and Vent System"), the stiffened pressure Suppression Chamber is a steel pressure vessel in the shape of a Torus. It is located below and encircles the Drywell, with a centerline diameter of approximately 111 ft and a cross-sectional diameter of 31 ft. It contains approximately 125,000 cu ft of water and has a gas space volume. The Suppression Chamber is supported on braced vertical columns to carry its dead and live loading to the reinforced concrete foundation slab of the reactor building.

2.3 Testing Requirements of 10 CFR 50, Appendix J, Option B

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, "Performance-Based Requirements." Amendment No. 214 for PBAPS, Unit 2 permits implementation of 10 CFR 50, Appendix J, Option B (Reference 2). TS 5.5.12 currently requires the establishment of a leakage rate testing program in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program implements the guidelines contained in Regulatory Guide (RG) 1.163 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, Option B, Section V.B.3 specifies that 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. 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 10 CFR 50, Appendix J, Option B.

The adoption of the Option B performance-based containment leakage rate testing program 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 containment leakage testing in Type A, B, and C tests. The required testing frequency is based upon an evaluation which utilizes the "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowable frequency for the Type A ILRT is based, in part, upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program" (Reference 3).

NUREG-1493 made the following findings with regard to changing the test frequency:

- Reducing the Type A ILRT 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 ILRTs 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 ILRTs have only been marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A ILRTs, increasing the interval between Type A ILRTs has minimal impact on public risk.

- While Type B and C tests identify the vast majority (i.e., greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

The required surveillance frequency for Type A ILRTs in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A ILRTs at least 24 months apart or refueling cycles where the calculated performance leakage rate was less than 1.0 L_a). Based on the ILRT history discussed below, the current test interval is 10 years.

3.0 TECHNICAL EVALUATION

3.1 10 CFR 50 Appendix J, Option B Plant Specific Implementation

As noted previously, License Amendment No. 214 implemented 10 CFR 50, Appendix J, Option B, for PBAPS, Unit 2 by adding TS Section 5.5.12. TS Section 5.5.12 requires Type A, B, and C testing in accordance with RG 1.163, which endorses the methodology for complying with Option B identified in NEI 94-01, Revision 0. The surveillance frequency for Type A testing in NEI 94-01 is at least once every 10 years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart demonstrate 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. The performance leakage rates are calculated in accordance with NEI 94-01, Section 9.1.1.

The two most recent Type A tests at PBAPS, Unit 2 have been satisfactory with leakage rates for the 1991 and 2000 Type A tests being 0.2135 wt%/day and 0.3365 wt%/day, respectively. These results are less than the acceptance criteria of 0.375 wt%/day, which was the acceptance criterion in effect at that time.

The specific results for the two most recent PBAPS, Unit 2 Type A ILRTs are as follows:

Year	wt%/day
1991	0.2135
2000	0.3365

The current limit for the maximum allowable Primary Containment leakage rate is 0.525 wt%/day at P_a (49.1 psig).

3.2 Plant Testing and Inspection Methods

In addition to periodic Type A testing, various inspections and tests are routinely performed to assure Primary Containment integrity. These include Type B and C testing performed in accordance with Appendix J, Option B; inspection activities performed as part of the American Society of Mechanical Engineers (ASME) Code Section XI (Subsection IWE) for the Primary Containment; Drywell IWE-2400 inspections; and Torus corrosion monitoring. Further discussion of these testing and inspection programs is provided below. The aggregate results of these tests and inspections provide a high degree of assurance of continued Primary Containment integrity.

3.3 Type B and Type C Testing Program

The PBAPS, Unit 2 Appendix J, Type B and Type C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the program as required by 10 CFR 50, Appendix J, Option B, and RG 1.163. The Type B and C test program consists of local leak rate testing of penetrations with a resilient seal, expansion bellows, double-gasketed manways, hatches and flanges, Drywell airlocks, and containment isolation valves that serve as a barrier to the release of the post-accident Primary Containment atmosphere. These components are tested at a pressure greater than 49.1 psig

(P_a). The results of the test program are used to ensure that proper maintenance and repairs are made on the Primary Containment components over their service life. The Type B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining the leakage from these components below appropriate limits.

As previously noted, Type B and Type C testing evaluates all but a small portion of potential containment leakage pathways. This License Amendment Request does not affect the scope, performance, or scheduling of Type B or Type C tests. Type B and Type C testing will continue to provide a high degree of assurance that Primary Containment integrity is maintained.

This License Amendment Request will not affect the scope, performance, or scheduling of Type B and C testing of containment penetrations and isolation valves. It is noted that Type B leak rate testing is on a maximum ten-year testing interval. If the result of a Type B test exceeds the allowable limit, the penetration will be returned to the short interval (every refueling outage). The Type C leak rate testing of containment isolation valves are on a maximum 60-month testing interval. If the result of a Type C test exceeds the allowable limit, the penetration will be returned to the short interval (every refueling outage).

In the Reference 7 letter, Constellation Energy requested a similar License Amendment Request. In the Reference 8 letter, Constellation Energy provided responses to the following NRC requests for additional information. The following is our response to these questions based on a review of historical Type B and C data:

<u>Question:</u>

a. *A summary list of those containment penetrations (including their test schedule intervals) that have not demonstrated acceptable performance history in accordance with the primary containment leakage rate program.*

<u>Response:</u>

See Enclosure 1.

<u>Question:</u>

b. *A summary table for Type B and Type C tests, including the interval schedule dates, that are planned to be performed prior to and during the requested five-year extension period of the ILRT interval.*

<u>Response:</u>

See Enclosure 2.

<u>Question:</u>

c. *Type B and Type C test results and their comparison with the allowable leakage rate specified in the plant Technical Specifications.*

Response:

Refueling Outage	Maximum Pathway		Minimum Pathway	
	Leakage (sccm)	% of 0.6La	Leakage (sccm)	% of 0.6La
2006 (2R16)	53,701	71.4	28,108	37.3
2008 (2R17)	59,033	56.0	20,281	19.3

Note:

0.6La prior to 2008 = 75,256 sccm
0.6La after implementation of Alternate Source Term in 2008 = 105,350 sccm (Safety Evaluation Report dated September 5, 2008)
sccm = standard cubic centimeters per minute

Question:

d. Testing and schedule of those penetrations with seals and gaskets, and bolted connections that are frequently disassembled or are not routinely disassembled.

Response:

See Enclosure 3.

3.4 Primary Containment Inspection Requirements — ASME Code Section XI, Subsection IWE

10 CFR Part 50, Appendix J, Option B, Section III.A, states: "A general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be conducted prior to each test, and at a periodic interval between tests based on the performance of the containment system." This inspection is also conducted during two other refueling outages before the next Type A test if the interval for the Type A test has been previously extended to 10 years, in order to allow for early discovery of structural deterioration.

Effective September 1996, the NRC amended 10 CFR 50.55a to endorse Subsections IWE and IWL of the ASME Code, Section XI, 1992 Edition including 1992 Addenda. These subsections contain inservice inspection (ISI) and repair/replacement rules for Class MC (metal containment) and Class CC (concrete containment) components. The PBAPS, Unit 2 Primary Containment is a freestanding steel containment, to which the requirements of Subsection IWE apply.

The Subsection IWE containment inspection requirements are implemented at PBAPS through the Containment Inservice Inspection (CISI) program. The program contains detailed inservice inspection requirements for Class MC components in accordance with 10 CFR 50.55a(b)(2)(vi) and (ix), and the ASME Code, Section XI, 1992 Edition with 1992 Addenda for the first CISI program interval. The second CISI program interval contains detailed inservice inspection requirements for Class MC components in accordance with 10 CFR 50.55a(b)(2)(vi) and (ix), and the ASME Code, Section XI, 2001 Edition with 2003 Addenda (Reference 4).

The general visual examination requirements specified in the IWE ISI program satisfy the visual examination requirements specified in Option B. The first Interval for the CISI Program became effective on November 5, 1998 and ended on November 4, 2008. The second Interval for the CISI Program became effective on November 5, 2008 and is scheduled to end on November 4, 2018. The three inspection periods for the first and second inspection intervals (as currently planned) are shown in the following table:

	1st 10-year interval	1st Period	2nd Period	3rd Period
Unit 2	11/5/98 - 11/4/08	11/5/98 - 11/4/02	11/5/02 - 11/4/05	11/5/05 - 11/4/08
Refuel Outages		2R13 – 2000 2R14 – 2002	2R15 – 2004	2R16 – 2006 2R17 – 2008
	2nd 10-year interval	1st Period	2nd Period	3rd Period
Unit 2	11/5/08 - 11/4/18	11/5/08 - 11/4/12	11/5/12 - 11/4/15	11/5/15 - 11/4/18
Refuel Outages		2R18 – 2010 2R19 – 2012	2R20 – 2014	2R21 – 2016 2R22 – 2018

Examinations are performed in accordance with non-destructive examination procedure MA-PB-793-001, "Visual Examination of Containment Vessels and Internals" and ST-N-080-900-2, "Visual Examination of Drywell and Torus Surfaces." These procedures provide the overall requirements and acceptance criteria for visual examinations in accordance with ASME Section XI, Article IWE, as delineated in the CISI program, in accordance with the ASME Code, Section XI, 1992 Edition with 1992 Addenda, and 2001 Edition with 2003 Addenda.

The corrosion rate criteria are based on maintaining a wall thickness of greater than the minimum design value. Acceptance criteria have been established and are documented in procedure MA-PB-793-001. Flaws identified during inspections are described as nicks, gouges, arc strikes, cracking, rust, or pitting. For each flaw, varying levels of severity are described and are evaluated as acceptable, unacceptable, or requiring further evaluation. Examples of criteria are provided below:

Drywell, Drywell Head	Light surface oxidation, discoloration
Torus Shell Corrosion	<0.0585"
Vents Corrosion	<0.0945"
Vent Header & Downcomers Corrosion	<0.1305"

Examination of pressure-retaining bolted connections and evaluation of containment bolting flaws or degradation will be performed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix)(G) and 10 CFR 50.55a(b)(2)(ix)(H).

Containment inspections will continue to be performed during the proposed 5-year extension of the Type A test interval (October 2010 through October 2015), in accordance with the CISI program. The CISI program requires a minimum of one inspection during each inspection period of the inspection interval. This extension will coincide with the first and second inspection periods of the second ten-year interval of the CISI program.

3.5 Drywell IWE-2400 Inspections

IWE examinations were performed in the first CISI interval in accordance with the 1992 Edition, 1992 Addenda of the ASME Section XI Code. These exams were performed in accordance with the ten-year frequency as defined by IWE-2400.

During the 2008 PBAPS refueling outage (2R17), a VT-3 examination of 100 percent of the accessible portions of the interior surface of the Drywell shell was performed. Two (2) recordable indications were identified and were entered into the corrective action program. The first recordable indication was that the sliding bolt at the 8 o'clock position on the Drywell equipment hatch was missing a cotter pin. The cotter pin was replaced and a satisfactory re-inspection was performed prior to the end of the refueling outage. The other recordable indication was that several small areas of the Drywell moisture barrier were peeling. The moisture barrier is located at the junction of the Drywell floor to Drywell wall. Conditions did not indicate the presence of any moisture accumulating on or behind the moisture barrier. Repairs were made to the areas and a satisfactory re-inspection was performed prior to the end of the refueling outage.

A General Visual (GV) examination of all accessible areas of the Unit 2 Drywell is next scheduled for 2010 (2R18) and 2014 (2R20). There are no IWE augmented inspections required for the Unit 2 Drywell.

PBAPS, Unit 2 implements a safety-related coatings program that ensures Design Bases Accident (DBA) qualified coating systems are used inside Primary Containment. The program assures that safety-related DBA qualified coatings (service level 1) are selected, procured, applied and inspected in a manner that conforms to the applicable 10 CFR 50 Appendix B criteria. Unqualified coatings are controlled and tracked to ensure that emergency core cooling systems will not be adversely affected by the coating debris following an accident. The program objective is to conform to licensee commitments made in response to Generic Letter 98-04. Safety-related coatings are also monitored in accordance with a formal Maintenance Rule (10 CFR 50.65) condition-monitoring program. Engineering reviews and evaluates the results of coating condition examinations performed by examiners qualified in accordance with ASTM D 4537, 1991 Edition.

3.6 Torus Corrosion Monitoring

The PBAPS Suppression Chamber (Torus) is constructed from carbon steel SA-516 Grade 70, material. The upper half, or vapor phase, of the shell was fabricated to a thickness of 0.604" and the lower half, or immersion phase, of the shell was fabricated to a thickness of 0.675". The internal design pressure of 62 psi dictates that the minimum global thickness of the Torus shell for the immersion phase must be equal to or greater than 0.599". To establish reasonable assurance that the minimum wall thickness of 0.599" is not reached for the immersion phase, PBAPS monitors Torus wall thickness and corrosion rate for the immersion phase. Determination of Torus corrosion rates is an ongoing activity that considers inspection results and the remaining corrosion allowance. There have been no areas of significant degradation requiring monitoring beyond the inspection requirements of ASME Section XI.

The PBAPS Torus Corrosion Monitoring Program has been developed to monitor the Torus shell immersion phase material thickness and ensure it is maintained within the bounds of the qualification bases. Assessment of observed Torus shell conditions ensures that timely action can be taken to correct degradation that could lead to loss of the intended function. The

program was based on a commitment to periodically monitor Torus condition as described in the NRC's Safety Evaluation Report dated September 17, 1999 (Reference 5) for the first CISI interval. The program focuses on condition monitoring in the form of component inspections and analysis, including the following elements:

- Periodic external Torus wall UT thickness measurements are obtained over a pre-defined grid system on the immersion phase.
- Diver inspection and measurement of pits on the immersion phase.
- Visual inspections of accessible external surfaces of the Torus support structure are performed. Included are inspections of the existing coatings of the base plates, anchor bolts, pipe columns, stiffener plates, tie rod cross braces, etc.

Monitoring in this manner ensures the Torus shell material will not be reduced to less than the minimum required wall thickness, and that any degradation is detected before there is a loss of intended function.

During the 2006 PBAPS, Unit 2 refueling outage (P2R16), a VT-1 examination of 100 percent of the pitting on the submerged portion of the Torus pressure boundary was performed. All pitting was measured. Pits with a depth of greater than 57 mils (52 mils for some thinner plates) were repaired with coating. Pits with a depth of greater then 67 mils (62 mils for some thinner plates) were evaluated by Engineering and repaired with coating. A VT-1 examination of 100 percent of the pitting on the submerged portion of the Torus pressure boundary is scheduled to be performed in 2010 (P2R18).

External Torus wall UT thickness measurements are next scheduled for 2010 (P2R18) in accordance with the IWE CISI program.

The most recent inspections of the Torus external structures, performed during the 2008 (P2R17) refueling outage, found the condition of the structures acceptable with no signs of missing or loose hardware, spalled concrete or major degradation that would impact the structural integrity of the Torus structure.

A General Visual (GV) examination of all accessible surfaces of the non-wetted portions of the Torus is next scheduled for 2010 (P2R18) and 2014 (P2R20).

The Torus Corrosion Monitoring Program is typically a more effective method than Appendix J Type A tests (ILRT) for identifying degrading minimum wall conditions, since the Type A test will only identify an actual breach in the pressure boundary.

3.7 Plant Operational Performance

As discussed in Section 5.2.3.2 of the PBAPS UFSAR ("Drywell"), the normal environment in the Drywell during plant operation is maintained at a slightly positive pressure and a bulk average ambient temperature of 145°F or lower. Temperature is maintained by recirculating the Drywell atmosphere across forced draft air cooling units which are cooled by the Drywell chilled water cooling system. Drywell temperature and pressure are continuously indicated in the main control room.

During power operation, the PBAPS, Unit 2 Primary Containment is inerted with nitrogen to maintain oxygen concentration within TS limits.

3.8 NRC Information Notice 92-20, Inadequate Local Leak Rate Testing

NRC Information Notice 92-20 was issued to alert licensees to problems with local leak rate testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem.

The bellows listed in UFSAR Table 5.2.2, "Containment Penetrations Compliance with 10CFR 50, Appendix J," are testable bellows and are tested in accordance with 10 CFR 50, Appendix J, Option B, Type B testing. Local leak rate test procedures for containment expansion bellows include verification of flow through the annulus between plies of the bellows, which ensures that restrictions between the plies that could conceal a leakage path do not exist.

3.9 Through-Wall Torus Shell Crack at James A. Fitzpatrick Nuclear Power Plant

A through-wall Torus shell crack was discovered at the James A. Fitzpatrick Nuclear Power Plant (JAF) on June 27, 2005. EGC reviewed the issue for applicability to PBAPS, and documented the results in the corrective action program.

The JAF High Pressure Coolant Injection (HPCI) turbine exhaust line that discharges into the suppression pool is open ended and does not have an end cap or a sparger. The PBAPS system configurations would not introduce the type of event that occurred at JAF. With respect to PBAPS, Unit 2, the HPCI system design does employ the use of a sparger on the turbine exhaust line. VT-2 and VT-3 inspections were performed on the nozzle and the Torus shell next to the HPCI and RCIC (Reactor Core Isolation Cooling) exhaust penetrations and the support legs to the Torus shell with satisfactory results. No further actions were required for PBAPS, Unit 2.

3.10 Generic Letter 87-05, Request for Additional Information - Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells

Generic Letter 87-05 described Drywell shell degradation, which occurred at Oyster Creek Nuclear Generating Station as a result of water intrusion into the air gap between the outer Drywell surface and the surrounding concrete and subsequent wetting of the sand cushion at the bottom of the air gap. The initial response to this generic letter for PBAPS was provided in a letter to the NRC dated May 11, 1987 (Reference 6).

PBAPS performs visual examinations for evidence of moisture in the air gap between the outer surface of the Drywell and surrounding concrete by opening each of the eight (8) stabilizer hatches in the upper elevation of the Drywell once per ten (10) years. No evidence of moisture has been found. Three (3) stabilizer hatches were opened during the most recent refueling outage in 2008 (P2R17).

The four (4) air gap drain lines are visually inspected in the Torus room for leakage when the cavity is flooded once per CISI inspection period. The most recent inspection was performed satisfactorily in 2008 (P2R17).

A functional test (i.e., smoke test) is performed once every cycle in accordance with procedure ST-N-007-900-2 to verify that the Drywell airgap drain lines are unclogged and functional. Per these procedures, the test verifies that the drain lines are free of water. The surveillance test was most recently performed on April 28, 2009 and the test was satisfactory.

3.11 Monitoring of Drywell Interior Coating

In addition to the inspections performed in accordance with the IWE CISI program, periodic visual inspections of the coating on accessible interior surfaces of the Drywell shell and Drywell head are performed to identify evidence of deterioration. The inspections are performed at a minimum every four years by surveillance test to identify any visible defects including blistering, cracking, flaking, peeling, and physical or mechanical damage. When degraded coatings are identified, evaluations are performed to determine any necessary actions (e.g., repair, removal, or replacement).

This surveillance test was last performed in 2008 (P2R17) with no areas identified as deteriorating. The next scheduled performances of this test are 2010 (P2R18) and 2014 (P2R20).

3.12 Moisture Barrier Inspection

The moisture barrier between the Drywell shell and the concrete floor at elevation 119' prevents moisture from entering the gap between the concrete and the Drywell shell and protects against corrosion of the shell in the inaccessible areas below the 119' level. The moisture barrier is inspected each CISI inspection period in accordance with the IWE CISI program. The most recent inspection was in 2008 during the P2R17 refueling outage. The inspection found some degradation of the moisture barrier sealant (as discussed in Section 3.5); however, no unacceptable degradation in the visible areas of the Drywell shell adjacent to the moisture barrier was found. The moisture barrier was repaired during the 2008 (P2R17) refueling outage by applying new sealant where required.

3.13 Plant Specific Risk Assessment

An evaluation was performed to assess the risk impact of a one-time extension of the PBAPS, Unit 2 containment ILRT interval from 10 years to 15 years. As discussed in this assessment (Attachment 4):

> "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 October 2008 EPRI final report [22]."

Additionally, Appendix A of Attachment 4 discusses compliance with Regulatory Guide 1.200.

The following conclusions regarding the assessment of the plant risk (Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)) are associated with extending the Type A ILRT test frequency to fifteen years as discussed in Section 7.0 of Attachment 4:

- Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in 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 estimated as 4.02E-08/yr using the NEI guidance as written, and at 1.47E-08/yr using the EPRI Expert Elicitation methodology. 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 for the base case with corrosion included is 4.51E-08. In both cases, the NEI guidance and the EPRI Expert Elicitation Methodology, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Regulatory Guide 1.174.

- 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 4.75E-02 person-rem/yr using the NEI guidance, and drops to 9.71E-03 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 1.03% using the NEI guidance, and drops to about 0.38% using the EPRI Expert Elicitation methodology. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.

- Since the increase in LERF falls well below the "small change" category using the acceptance guidelines of Regulatory Guide 1.174, a detailed examination of the external events impact is not explicitly required, nor is it expected to change the conclusions from this assessment.

- To confirm the expected impact from external events, an additional bounding assessment of the potential impact from the risk associated with external events was done. As shown in Table 5.7-1, the total increase in LERF due to internal events and the bounding external events assessment is 5.50E-07/yr, which is in Region II of the Regulatory Guide 1.174 acceptance guidelines.

- Finally, the same bounding analysis indicates that the total LERF from internal and external risks as shown in Table 5.7-2 is 2.75E-06/yr, which is less than the Regulatory Guide 1.174 limit of 1E-05/yr given that the ΔLERF is in Region II.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the PBAPS, Unit 2 risk profile.

3.14 Summary

Based on the previous ILRT tests conducted at PBAPS, Unit 2, which confirm that the Primary Containment structure exhibits extremely low leakage, EGC concludes that the one-time extension of the containment ILRT interval from 10 to 15 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and C testing performed in accordance with Option B of 10 CFR 50 Appendix J, inspection activities performed as part of the plant IWE ISI program, the Torus corrosion monitoring program, inspections of Drywell interior coatings, and by operating experience with a containment that normally operates at a positive pressure (i.e., the pressure from containment inerting). In the aggregate, these provide continuing confidence in containment integrity.

This experience is supplemented by risk analysis studies, including the PBAPS, Unit 2 risk analysis provided in Attachment 4. The findings of the risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT test interval from 10 years to 15 years results in a very small change to the PBAPS, Unit 2 risk profile.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR Part 50, Appendix J, Option B, requires that licensees' primary reactor containments meet the leakage rate requirements as delineated by Appendix J. This requirement is met by performance of Type A, B, and C leakage rate testing on the Primary Containment and its associated components (e.g., valves, penetrations). The leakage rate test results are compared to allowable leakage rate acceptance criteria set forth by Appendix J. PBAPS, Unit 2 TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," invokes Appendix J requirements.

TS Section 5.5.12 requires that the leakage rate testing of the containment be performed in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, and in accordance with NRC Regulatory Guide 1.163 dated September 1995 with NRC-approved exceptions. Regulatory Position C.1 of Regulatory Guide 1.163 states that licensees should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01. Section 11.0 of NEI 94-01 references Section 9.0, which would require that ILRTs be performed within 10 years from the date of their last performance. PBAPS, Unit 2 will continue to comply with the requirements of 10 CFR 50, Appendix J, with the proposed ILRT extension. No other regulations or TS are affected by the proposed amendment.

4.2 Precedents

The NRC has approved similar risk-informed license amendment requests relating to a one-time extension of the ILRT interval for a number of plants:

- Peach Bottom Atomic Power Station, Unit 3 (License Amendment No. 244 issued by NRC letter date October 4, 2001 – TAC No. MB2094).

- Vermont Yankee Nuclear Power Station (License Amendment No. 227 issued by NRC letter dated August 31, 2005 - TAC No. MC4662).

- Cooper Nuclear Station (License Amendment No. 224 issued by NRC letter dated October 3, 2006 - TAC No. MC9732).

- Nine Mile Point Nuclear Station, Unit 1 (License Amendment No. 202 issued by NRC letter dated March 11, 2009 – TAC No. MD9453).

4.3 No Significant Hazards Consideration

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

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

   Response: No

   The proposed change involves a one-time extension of the Primary Containment ILRT interval from 10 years to 15 years. The proposed change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The Primary Containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed change.

   Continued containment integrity is assured by the established programs for local leak rate testing and inservice/containment inspections, which are unaffected by the proposed change. As documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995, industry experience has shown that local leak rate tests (Type B and C) have identified the vast majority of containment leakage paths, and that ILRTs detect only a small fraction of containment leakage pathways.

   The potential consequences of the proposed change have been quantified by analyzing the changes in risk that would result from extending the ILRT interval from 10 years to 15 years. Increasing the ILRT interval to 15 years for this one-time change is considered to be insignificant since it represents a very small change to the PBAPS, Unit 2 risk profile. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. PBAPS, Unit 2 has determined that the increase in conditional containment failure probability due to the proposed change is very small. Therefore, it is concluded that the proposed one-time extension of the Primary Containment ILRT interval from 10 years to 15 years does not significantly increase the consequences of an accident previously evaluated.

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

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

   Response: No

   The proposed change involves a one-time extension of the Primary Containment ILRT interval. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change in the manner in which the plant is operated or controlled.

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

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

   Response: No

   The proposed one-time extension of the Primary Containment ILRT interval does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the 10 CFR 50 Appendix J testing program plan, as defined in the Technical Specifications, exist to ensure that the degree of Primary Containment structural integrity and leak-tightness that is considered in the plant safety analyses is maintained. The overall containment leakage rate limit specified by the Technical Specifications is maintained, and Type B and C containment leakage tests will continue to be performed at the frequency currently required by the TS.

   Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by an ILRT. Furthermore, a risk assessment using the current PBAPS, Unit 2 Probabilistic Risk Assessment internal events model concluded that extending the ILRT test interval from 10 years to 15 years results in a very small change to the PBAPS, Unit 2 risk profile.

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

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

4.4 Conclusions

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

and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

(1) Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995

(2) Letter from J. Shea (U.S. Nuclear Regulatory Commission) to G. A. Hunger (PECO Energy Company, LLC), "Technical Specifications Regarding 10 CFR 50, Appendix J, Option B, Containment Leakage Rate Testing Requirements, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3 (TAC NOS. M94853 and M94854)," dated June 18, 1996

(3) NUREG-1493, "Performance-Based Containment Leak-Test Program," dated July 1995

(4) Letter from H. Chernoff (U.S. Nuclear Regulatory Commission) to C. Pardee (Exelon Generation Company, LLC), "Peach Bottom Atomic Power Station, Units 2 and 3 – Requests for Relief Associated with the Third and Fourth Inservice Testing Intervals and the First and Second Containment Inservice Intervals (TAC NOS. MD8294, MD8295, MD8296, MD8297, MD8298, MD8299, MD8300, MD8301, MD8302, MD8303, MD8304, MD8305, MD8306, MD8307, MD8308, and MD8309)," dated February 26, 2009

(5) Letter from J. Clifford (U.S. Nuclear Regulatory Commission) to J. Hutton (PECO Energy Company), "Safety Evaluation for Proposed Alternatives to ASME Section XI Requirements for Containment Inservice Inspection, Peach Bottom Atomic Power Station, Units 2 and 3 (TAC NOS. MA4973 and MA4974)," dated September 17, 1999

(6) Letter from J. Gallagher (Philadelphia Electric Company) to S. Varga (U.S. Nuclear Regulatory Commission), "Peach Bottom Atomic Power Station Generic Letter 87-05 dated March 12, 1987 Degradation of Mark I Drywells," dated May 11, 1987

(7) Letter from K. Polson (Constellation Energy) to U.S. Nuclear Regulatory Commission, "License Amendment Request Pursuant to 10 CFR 50.90: One-Time Extension of the Primary Containment Integrated Leakage Rate Test Interval – Technical Specification Section 6.5.7, 10 CFR 50 Appendix J Testing Program Plan," dated August 15, 2008

(8) Letter from K. Polson (Constellation Energy) to U.S. Nuclear Regulatory Commission, "License Amendment Request Pursuant to 10 CFR 50.90: One-Time Extension of the Primary Containment Integrated Leakage Rate Test Interval – Response to NRC Request for Additional Information (TAC No. MD9453)," dated December 4, 2008

Enclosure 1

Summary List of Containment Penetrations With Unacceptable Performance History

Summary List of Containment Penetrations With Unacceptable Performance History

Summary list of those containment penetrations that have not demonstrated acceptable performance history in accordance with the Primary Containment leak rate test program since the last Primary Containment integrated leak rate test. These tables reflect the testing performed through the operating cycle and refueling outage.

2002 Fall Refueling Outage and Operating Cycle

Pen.#	Component ID	System	Test Type	Admin. Limit	Test Result (SCCM)	Comments
9B	CHK-2-06-96B	Feedwater	C	9000	26175	Exceeded admin. limits / valve repaired / 30 months fixed / not Option B qualified (cannot be extended)
205B	AO-2-07B-2502A	Containment	C	7500	off scale	Exceeded admin. limits / valve repaired / 30 months fixed / not Option B qualified (cannot be extended)
25	CHK-07B-40095B	Containment	C	500	862	Exceeded admin. limits / valve repaired / return to 30 month frequency
11	MO-2-23-15, MO-2-23-16	HPCI	C	3000	off scale	Exceeded admin limits / valve repaired / return to 30 month frequency

2004 Fall Refueling Outage and Operating Cycle

Pen. #	Component ID	System	Test Type	Admin. Limit	Test Result (SCCM)	Comments
51B	SV-2-07D-2978D	Containment	C	100	406	Exceeded admin. limits / valve repaired / return to 30 month frequency
39B	MO-2-10-26A,MO-2-10-31A	RHR	C	5000	7750	Exceeded admin. limits / valve repaired / return to 30 month frequency
13A	AO-2-10-46B	RHR	C	7000	Off scale	Exceeded admin. limits / valve repaired / return to 30 month frequency
13A	AO-2-10-163B	RHR	C	250	449	Exceeded admin. limits / valve repaired / return to 30 month frequency
10	MO-2-13-15, MO-2-13-16	RCIC	C	1000	2784	Exceeded admin. limits / valve repaired / return to 30 month frequency
16B	MO-2-14-012A	Core Spray	C	1500	1547	Exceeded admin. limits / valve repaired / return to 30 month frequency
16A	AO-2-14-15B	Core Spray	C	250	985	Exceeded admin. limits / valve repaired / return to 30 month frequency
19	AO-2-20-95	Radwaste	C	1000	10145	Exceeded admin. limits / valve repaired / return to 30 month frequency
217B	CHK-2-23-65	HPCI	C	5000	21400	Exceeded admin. limits / valve repaired / 30 months fixed / not Option B qualified (cannot be extended)

Summary List of Containment Penetrations With Unacceptable Performance History

2006 Fall Refueling Outage and Operating Cycle

Pen. #	Component ID	System	Test Type	Admin. Limit	Test Result (SCCM)	Comments
8	MO-2-01A-74, MO-2-01A-77	Main Steam	C	1500	7182	Exceeded admin. limits / valve repaired / return to 30 month frequency
9B	MO-06-38B	Feedwater	C	7000	off scale	Exceeded admin. limits / valve repaired / 30 months fixed / not Option B qualified (cannot be extended)
110A -H	RPV STABILIZER MANWAYS	Containment	B	800	862	Exceeded admin. limits / valve repaired / return to 30 month frequency
201E F	201E&F D/W TO TORUS EXPAN JOINT	Containment	B	250	3000	Exceeded admin. limits / valve repaired / return to 30 month frequency
51C	SV-2-07D-2671C	Containment	C	100	off scale	Exceeded admin. limits / valve repaired / return to 30 month frequency
51D	CHK-2-07D-40140	Containment	C	500	off scale	Exceeded admin. limits / valve repaired / return to 30 month frequency
39B	MO-2-10-26A	RHR	C	5000	8660	Exceeded admin. limits / valve repaired / return to 30 month frequency
211A	MO-2-10-34B, 38B, 39B	RHR	C	7000	11706	Exceeded admin. limits / valve repaired / return to 30 month frequency
13A	AO-2-10-46B	RHR	C	7000	off scale	Exceeded admin. limits / valve repaired / return to 30 month frequency
218A	AO-2-16-2968	Instr. N_2	C	250	903	Exceeded admin. limits / valve repaired / return to 30 month frequency
19	AO-2-20-94	Radwaste	C	1000	1250	Exceeded admin. limits / valve repaired / return to 30 month frequency
19	AO-2-20-95	Radwaste	C	1000	1930	Exceeded admin. limits / valve repaired / return to 30 month frequency

2008 Fall Refueling Outage and Operating Cycle

Pen. #	Component ID	System	Test Type	Admin. Limit	Test Result (SCCM)	Comments
8	MSL Drain MO-2-01A-74, MO-2-01A-77	Main Steam	C	1500	4496	Exceeded admin. limits / valve repaired / return to 30 month frequency
203	SV-2-07D-2671B	Containment	C	100	off scale	Exceeded admin. limits / valve repaired / return to 30 month frequency
51C	SV-2-07D-2671C	Containment	C	100	off scale	Exceeded admin. limits / valve repaired / return to 30 month frequency
51A	SV-2-07D-2671E	Containment	C	100	320	Exceeded admin. limits / valve repaired / return to 30 month frequency
26	SV-2-07D-2978G	Containment	C	100	740	Exceeded admin. limits / valve repaired / return to 30 month frequency
51D	SV-2-07D-2980	Containment	C	100	183	Exceeded admin. limits / valve repaired / return to 30 month frequency

Summary List of Containment Penetrations With Unacceptable Performance History

2008 Fall Refueling Outage and Operating Cycle (continued)

Pen. #	Component ID	System	Test Type	Admin. Limit	Test Result (SCCM)	Comments
39B	MO-2-10-26A	RHR	C	5000	19520	Exceeded admin. limits / valve repaired / return to 30 month frequency
39A	MO-2-10-26B	RHR	C	5000	off scale	Exceeded admin. limits / valve repaired / return to 30 month frequency
13A	AO-2-10-46B	RHR	C	7000	7770	Exceeded admin. limits / valve repaired / return to 30 month frequency
18	AO-2-20-82	Radwaste	C	1000	off scale	Exceeded admin. limits / valve repaired / return to 30 month frequency

Enclosure 2

Summary Table for Type B and Type C Tests Planned to be Performed During the ILRT Extension

Summary Table for Type B and Type C Tests Planned to be Performed During the ILRT Extension

Summary table for Type B and Type C tests, including the interval schedule dates that are planned to be performed prior to and during the requested 5-year extension period of the ILRT interval. We note that current testing may impact the future test schedule. As an example, a penetration failure may cause the next scheduled test to occur earlier.

Pen. #	Component	Last Test	Through 2R18 Tested 2010	Through 2R19 Tested 2012	Through 2R20 Tested 2014
7A	MSIV'S AO-80A, AO-86A	9/16/08	YES	YES	YES
7B	MSIV'S AO-80B, AO-86B	9/16/08	YES	YES	YES
7C	MSIV'S AO-80C, AO-86C	9/19/08	YES	YES	YES
7D	MSIV'S AO-80D, AO-86D	9/16/08	YES	YES	YES
8	MSL Drain MO-2-01A-74, MO-2-01A-77	9/16/08	YES	YES	YES
57	AO-2-02-316, AO-2-02-317	9/26/06	NO	YES	NO
41	AO-2-02-039, AO-2-02-040	9/29/06	NO	YES	NO
9A	CHK-2-06-28A	9/19/08	YES	YES	YES
9A	CHK-2-06-96A, MO-2-06-038A, MO-2-23-019	9/19/08	YES	YES	YES
9B	CHK-2-06-28B	9/20/08	YES	YES	YES
9B	CHK-2-06-96B, MO-2-06-038B, MO-2-13-021, MO-2-12-068	9/20/08	YES	YES	YES
2	PERSONNEL AIRLOCK	10/20/08	YES	YES	YES
DW/HD	DW HEAD SEAL	10/6/08	YES	YES	YES
6	CRD HATCH	10/7/08	YES	YES	YES
200A	N/E HATCH	10/1/08	YES	YES	YES
200B	S/W HATCH	9/28/08	YES	YES	YES
1	D/W EQUIPMENT ACCESS HATCH	10/4/08	YES	YES	YES
2	D/W AIRLOCK O-RING	9/20/08	NO	NO	YES
35	TIP O-RING	9/23/08	NO	NO	YES
4	D/W HEAD ACCESS	9/30/06	NO	YES	NO
RPV/STA	RPV STABILIZER MANWAYS	9/16/08	YES	YES	YES
30S199	E20S199 (Electrical Panel)	9/17/06	NO	YES	NO
30S198	E20S198 (Electrical Panel)	9/17/06	NO	YES	NO
30S200	E20S200 (Electrical Panel)	9/17/06	NO	YES	NO
201AB	201A&B D/W TO TORUS EXPAN JOINT	9/20/04	YES	NO	NO
201CD	201C&D D/W TO TORUS EXPAN JOINT	9/21/04	YES	NO	NO
201EF	201E&F D/W TO TORUS EXPAN JOINT	9/20/08	YES	YES	NO
201GH	201G&H D/W TO TORUS EXPAN JOINT	9/23/06	NO	YES	NO
30S199	EXPANSION 20S199	9/20/04	YES	NO	NO
30S198	EXPANSION 20S198	9/24/06	NO	YES	NO
30S200	EXPANSION 20S200	9/24/06	NO	YES	NO
150	N-150 TEST NOZZLE	9/19/04	YES	NO	NO
250	N-250 TEST NOZZLE	9/23/06	NO	YES	NO
32CD218	HV-2-07A-29871,73,75 HV-2-07A-29872,74,76	9/23/04	YES	NO	NO
205B	AO-2-07B-2502A, VBV-2-07B-26A TORUS VAC BKR	10/1/08	YES	YES	YES
205B	AO-2-07B-2502A O-RING	9/26/06	NO	YES	NO
205A	AO-2-07B-2502B, VBV-2-07B-26B TORUS VAC BKR	10/1/08	YES	YES	YES
205A	AO-2-07B-2502B O-RING	9/14/04	YES	NO	NO
26	AO-2-07B-2506, AO-2-07B-2507	9/24/08	YES	YES	YES
26	AO-2-07B-2506 O- RING	9/23/08	NO	NO	YES
26	AO-2-07B-2509, AO-2-07B-2510,	9/22/08	YES	YES	YES

Summary Table for Type B and Type C Tests Planned to be Performed During the ILRT Extension

Pen. #	Component	Last Test	Through 2R18 Tested 2010	Through 2R19 Tested 2012	Through 2R20 Tested 2014
	SV-2-16-8100, AO-2-16-4235				
219	AO-2-07B-2511, AO-2-07B-2512, AO-2-07B-80290	9/21/08	YES	YES	YES
219	AO-2-07B-2511 O- RING	9/21/08	NO	NO	YES
219	AO-2-07B-2513, 2514	9/17/08	YES	YES	YES
25	AO-2-07B-2505, 2519, 2520, 2521A&B	10/9/08	YES	YES	YES
25	AO-2-07B-2520 O-RING	9/22/04	YES	NO	NO
205B	AO-2-07B-2521B O-RING	9/14/08	NO	NO	YES
25	AO-2-07B-2523, CHK-2-07B-40095A&B	7/28/08	NO	YES	NO
39A	CHK-2-07C-40142, SV-2-07C-4949A, CAD INJ	9/21/08	YES	YES	YES
39B	CHK-2-07C-40143, SV-2-07C-4949B, CAD INJ	8/5/08	YES	YES	YES
211A	CHK-2-07C-40144, SV-2-07C-4951A, CAD INJ	7/23/08	YES	YES	YES
211B	CHK-2-07C-40145, SV-2-07C-4951B, CAD INJ	8/12/08	YES	YES	YES
218B	SV-2-07D-2671A, 2978A, 02 ANAL	8/25/08	YES	YES	YES
203	SV-2-07D-2671B, 2978B, 02 ANAL	8/27/08	YES	YES	YES
51C	SV-2-07D-2671C, 2978C, 02 ANAL	5/21/08	YES	YES	YES
51B	SV-2-07D-2671D, 2978D, 02 ANAL	8/11/08	YES	YES	YES
51A	SV-2-07D-2671E, 2978E, 02 ANAL	5/20/08	YES	YES	YES
219	SV-2-07D-2671F, 2978F, 02 ANAL	8/26/08	YES	YES	YES
26	SV-2-07D-2671G, 2978G, 02 ANAL	8/19/08	YES	YES	YES
51D	CHK-2-07D-40140, SV-2-07D-2980	8/13/08	YES	YES	YES
219	SV-2-07E-4960A, 4961A, SV-2-63G-4966A	9/3/08	NO	YES	NO
26	SV-2-07E-4960B, 4961B, SV-2-63G-4966B, 8101	8/14/08	YES	YES	YES
51C	SV-2-07E-4960C, 4961C, SV-2-63G-4966C	9/2/08	NO	YES	NO
203	SV-2-07E-4960D, 4961D, SV-2-63G-4966D	7/5/05	YES	NO	YES
12	MO-2-10-018, MO-2-10-017, S/D COOLING	9/28/06	NO	YES	NO
211B	MO-2-10-034A, 38A, 39A, TORUS COOL/SPRAY	9/26/08	YES	NO	NO
211B	MO-2-10-38A PACKING	9/26/08	NO	NO	YES
39B	MO-2-10-026A, MO-2-10-031A	10/7/08	YES	YES	YES
39B	MO-2-10-031A PACKING	9/15/04	YES	NO	NO
13B	AO-2-10-046A, AO-2-10-163A	9/27/08	YES	YES	YES
13B	MO-2-10-025A	9/27/08	YES	YES	YES
211A	MO-2-10-034B, 38B, 39B, TORUS COOL/SPRAY	9/16/08	YES	NO	NO
211A	MO-2-10-038B PACKING	10/3/06	NO	YES	NO
39A	MO-2-10-026B, MO-2-10-031B	9/20/08	YES	YES	YES
39A	MO-2-10-031B PACKING	9/26/06	NO	YES	NO
13A	AO-2-10-046B, AO-2-10-163B	9/17/08	YES	YES	YES
13A	MO-2-10-025B	9/17/08	YES	YES	YES
42	CHK-2-11-16, XV-2-11-14A&B	9/22/08	YES	YES	YES
14	MO-2-12-015, 18 RWCU SUCTION	9/20/04	YES	NO	NO
10	MO-2-13-015, MO-2-13-016, RCIC STM SUPPLY	9/20/08	NO	NO	YES
217B	CHK-2-13C-50	9/21/08	YES	YES	YES
217B	HV-2-13C-9 O-RING	9/16/06	NO	YES	NO
217B	AO-2-13-138	9/28/08	NO	NO	YES
217B	MO-2-13C-4244, External On HV-2-13C-21201 and HV-2-13C-21202	9/21/08	NO	NO	YES
16B	AO-2-14-013A, AO-2-14-015A	9/26/08	NO	NO	YES

Summary Table for Type B and Type C Tests Planned to be Performed During the ILRT Extension

Pen. #	Component	Last Test	Through 2R18 Tested 2010	Through 2R19 Tested 2012	Through 2R20 Tested 2014
16B	MO-2-14-012A	9/26/08	NO	NO	YES
16A	AO-2-14-013B, AO-2-14-015B	9/18/08	NO	NO	YES
16A	MO-2-14-012B	9/18/08	NO	NO	YES
22	CHK-2-16-23202A, AO-2-16-2969A	9/17/04	YES	NO	NO
52F	AO-2-16-2969B, CHK-2-16-23335	9/23/04	YES	NO	NO
52F	CHK-2-16-23202B, HV-2-16-23333	9/23/04	YES	NO	NO
218A	CHK-2-16-23261, AO-2-16-2968	9/22/08	YES	NO	NO
102BD	CHK-2-16A-23299A; SV-2-16A-8130A	8/30/05	YES	NO	YES
47	CHK-2-16A-23299B; SV-2-16A-8130B	10/28/05	YES	NO	YES
18	AO-2-20-082, AO-2-20-083	10/4/08	YES	YES	YES
19	AO-2-20-94, AO-2-20-95	9/27/08	YES	YES	YES
11	MO-2-23-015, MO-2-23-016 HPCI STM SUPPLY	9/28/08	YES	YES	NO
217B	CHK-2-23C-65	9/16/08	YES	YES	YES
217B	HV-2-23C-12 O-RING	9/21/08	NO	NO	YES
217B	AO-2-23-137	9/22/08	YES	NO	NO
217B	MO-2-23B-4245	9/27/08	NO	NO	YES
217B	MO-2-23B-4245, HV-2-23C-21122 PACKINGS	9/20/04	YES	NO	NO
23,24	MO-2-35-2373, 2374 RBCCW SUPPLY	9/26/04	YES	NO	NO
21	HV-2-36A-20163, 20165 D/W SERVICE AIR	10/4/06	NO	YES	NO
53	MO-2-44A-2201B	9/21/04	YES	NO	NO
54	MO-2-44A-2200B	9/24/04	YES	NO	NO
55	MO-2-44A-2200A	9/24/04	YES	NO	NO
56	MO-2-44 A-2201A	9/21/04	YES	NO	NO
35D	SV-2-07-109, CHK-2-07F-41504	8/20/08	NO	YES	NO
35BDEFG	TIP BALL VALVES	10/8/08	YES	YES	YES
206AB	TORUS LEVEL INDICATION	9/27/06	NO	YES	YES

Enclosure 3

Testing and Schedule of Penetrations with Seals, Gaskets, and Bolted Connections Frequently Disassembled or are Not Routinely Disassembled

Testing and Schedule of Penetrations with Seals, Gaskets, and Bolted Connections Frequently Disassembled or are Not Routinely Disassembled

The testing and schedule of those penetrations with seals, gaskets, and bolted connections that are frequently disassembled or are not routinely disassembled.

PBAPS Frequently Disassembled Gaskets / Bolted Penetrations
Type B test schedule

Pen. #	Component	Test Schedule Interval	Last Test
2	PERSONNEL AIRLOCK	30 month fixed	10/20/08
DW/HD	DW HEAD SEAL	30 month performance-based	10/6/08
6	CRD HATCH	30 month performance-based	10/7/08
200A	N/W HATCH	30 month performance-based	10/1/08
200B	S/W HATCH	30 month performance-based	9/28/08
1	D/W EQUIPMENT ACCESS HATCH	30 month performance-based	10/4/08
RPV/STA	RPV STABILIZER MANWAYS	30 month performance-based	9/16/08

PBAPS Infrequently Disassembled Gasketed / Bolted Penetrations
Type B test schedule

Pen. #	Component	Test Schedule Interval	Last Test
2	D/W AIRLOCK O-RING	75 months	9/20/08
35	TIP O-RING	75 months	9/23/08
4	D/W HEAD ACCESS	75 months	9/30/06
30S199	E20S199 (Electrical Panel)	75 months	9/17/06
30S198	E20S198 (Electrical Panel)	75 months	9/17/06
30S200	E20S200 (Electrical Panel)	75 months	9/17/06
201AB	201A&B D/W TO TORUS EXPAN JOINT	75 months	9/20/04
201CD	201C&D D/W TO TORUS EXPAN JOINT	75 months	9/21/04
201EF	201E&F D/W TO TORUS EXPAN JOINT	75 months	9/20/08
201GH	201G&H D/W TO TORUS EXPAN JOINT	75 months	9/23/06
30S199	EXPANSION 20S199	75 months	9/20/04
30S198	EXPANSION 20S198	75 months	9/24/06
30S200	EXPANSION 20S200	75 months	9/24/06
205B	AO-2-07B-2502A O-RING	75 months	9/26/06
205A	AO-2-07B-2502B O-RING	75 months	9/14/04
26	AO-2-07B-2506 O-RING	75 months	9/23/08
219	AO-2-07B-2511 O-RING	75 months	9/21/08
25	AO2-07B-2520 O-RING	75 months	9/22/04
205B	AO-2-07B-2521B O-RING	75 months	9/14/08
217B	HV-2-13-09 O-RING	75 months	9/16/06
217B	HV-2-23-12 O-RING	75 months	9/21/08
150	N-150 TEST NOZZLE	75 months	9/19/04
211B	MO-2-10-38A PACKING	75 months	9/26/08
39B	MO-2-10-31A PACKING	75 months	9/15/04
211A	MO-2-10-38B PACKING	75 months	10/3/06

Testing and Schedule of Penetrations with Seals, Gaskets, and Bolted Connections Frequently Disassembled or are Not Routinely Disassembled

Pen. #	Component	Test Schedule Interval	Last Test
39A	MO-2-10-31B PACKING	75 months	9/26/06
217B	MO-2-23-4245, HV-2-23-21122 PACKING	75 months	9/20/04
250	N-250 TEST NOZZLE	75 months	9/23/06

ATTACHMENT 2

MARKUP OF PROPOSED TECHNICAL SPECIFICATION PAGE CHANGE

Revised TS Pages

5.0-17 (Unit 2)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

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

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

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. 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.

5.5.12 Primary 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, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Section 10.2:

a. MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

b. Section 9.2.3: The first Type A test performed after the October 2000 Type A test shall be performed no later than October 2015.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.7% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is ≤ 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 Type C tests and ≤ 0.75 L_a for Type A tests;

(continued)

ATTACHMENT 3

RETYPED PAGE FOR TECHNICAL SPECIFICATION CHANGE

Retyped TS Pages

5.0-17 (Unit 2)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

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

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

3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.

c. 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.

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a. Section 10.2: MSIV leakage is excluded from the combined total of 0.6 L_a for the Type B and C tests.

b. Section 9.2.3: The first Type A test performed after the October 2000 Type A test shall be performed no later than October 2015.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.7% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

a. Primary Containment leakage rate acceptance criterion is ≤ 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 Type C tests and ≤ 0.75 L_a for Type A tests;

(continued)

ATTACHMENT 4

Risk Assessment for Peach Bottom Unit 2 to Support ILRT (Type A) Interval Extension Request

RM DOCUMENTATION NO.	PB-LAR-03	REV:	1	Page No.	1

STATION: Peach Bottom Atomic Power Station

UNIT(S) AFFECTED: Unit 2

TITLE: Risk Assessment for Peach Bottom Unit 2 to Support ILRT (Type A) Interval Extension Request

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

[ ] Review required after periodic update

[X] Internal RM Documentation

Electronic Calculation Data Files:

Microsoft Excel File PBu2_ILRT-Final.xls, 3/31/2009, 6:12 PM, 187 KB

	Name(s)	Signature	Date
Prepared by:	Shikha Prasad		5/15/09
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Reviewed by:	Robert J. Wolfgang	Robert J. Wolfgang	5/15/09
Approved by:	Jeff R. Gabor		5/15/09

Method of Review: [X] Detailed [ ] Alternate [ ] Review of External Document

This RM documentation supersedes: Rev. 0 **in its entirety.**

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

TABLE OF CONTENTS

List of Tables

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 Peach Bottom Atomic Power Station, Unit 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 October 2008 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 1.0La (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 for a comparable BWR plant, that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis is being performed to confirm these conclusions based on Peach Bottom Unit 2 specific models and available data. As defined by Plant Technical Specifications, the maximum primary containment leakage rate, La, is actually 0.7% of primary containment air weight per day at design basis accident pressure [30].

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 Peach Bottom Unit 2 employs the NEI Interim Guidance methodology, with the affected System, Structure, or Component (SSC) being the primary containment boundary.

1.3 ACCEPTANCE 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 for Peach Bottom Unit 2, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year provided that the total from all contributors (including external events) can be reasonably shown to be less than 10^{-5} 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 (refer to Table 2-1 and Table 2-2) from the current Peach Bottom Unit 2 PRA model and subsequent containment response for the various fission product release categories including no or negligible release (refer to Table 2-2 and Table 2-3).

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 Peach Bottom Unit 2 assessment uses population dose as one of the risk measures. The other risk measures used in the Peach Bottom Unit 2 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 Peach Bottom Unit 2 uses ground rules and methods to calculate changes in risk metrics that are similar to those used in the EPRI approach.

TABLE 2-1
SUMMARY OF THE PEACH BOTTOM UNIT 2 CORE DAMAGE FREQUENCY BY ACCIDENT SEQUENCE SUBCLASS

ACCIDENT CLASS DESIGNATOR	SUBCLASS	DEFINITION	PB UNIT 2 FREQUENCY (PER YEAR)
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	1.38E-06
	B	Accident sequences involving a loss of offsite power and loss of coolant inventory makeup.	1.26E-06
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	2.10E-09
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.	4.95E-08
	E	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high and dc power is unavailable.	5.07E-09
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage; core damage induced post containment failure.	3.30E-07
	F	Class IIA and IIL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	4.83E-07
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage induced post containment failure.	1.27E-08
Class III (LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	9.00E-09
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	4.55E-08
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	6.54E-08

TABLE 2-1
SUMMARY OF THE PEACH BOTTOM UNIT 2 CORE DAMAGE FREQUENCY BY ACCIDENT SEQUENCE SUBCLASS

ACCIDENT CLASS DESIGNATOR	SUBCLASS	DEFINITION	PB UNIT 2 FREQUENCY (PER YEAR)
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	4.34E-08
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	1.24E-07
	L	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially breached; core damage induced post containment failure.	5.36E-09
Class V	---	Unisolated LOCA outside containment.	9.42E-08
Total	---		3.91E-06

TABLE 2-2
SUMMARY OF THE PEACH BOTTOM UNIT 2 LEVEL 2 MODEL RESULTS BY RELEASE CATEGORY

RELEASE CATEGORY [1]	PB UNIT 2 FREQUENCY (PER YEAR)
M/I	7.81E-08
L/L	4.79E-07
M/L	9.79E-07
LLE	0.00E+00
L/I	3.96E-08
M/E	6.85E-07
LLL	3.13E-08
H/E	1.69E-07
L/E	0.00E-00
H/L	6.21E-08
H/I	5.50E-08
LLI	7.36E-09
Total[2]:	2.59E-06

(1) Refer to Table 2-3 for the release category classification scheme used in the Peach Bottom Unit 2 Level 2 analysis.

(2) The difference between this value and the total CDF value of 3.91E-06 is assigned to the Containment Intact (OK) category.

TABLE 2-3
RELEASE SEVERITY AND TIMING CLASSIFICATION SCHEME (SEVERITY, TIMING)

RELEASE SEVERITY TERM RELEASE FRACTION		RELEASE TIMING	
CLASSIFICATION CATEGORY	CS IODIDE % IN RELEASE	CLASSIFICATION CATEGORY	TIME OF RELEASE[1]
High (H)	Greater than 10	Late (L)	Greater than 24 hours
Moderate (M)	1 to 10	Intermediate (I)	6 to 24 hours
Low (L)	0.1 to 1	Early (E)	Less than 6 hours
Low-low (LL)	Less than 0.1		
No Iodine (OK)	0		

(1) Relative to the declaration of a General Emergency.

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The Peach Bottom Unit 2 Level 1 and Level 2 internal events PRA models provide representative results.
- It is appropriate to use the Peach Bottom Unit 2 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; however fire and seismic events have been accounted for in the analysis.
- 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 the more recent population for Peach Bottom Unit 2 compared to the population estimates for Peach Bottom used in 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 1La. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is 10La, based on the previously approved methodology performed for Indian Point Unit 3 [6, 7].
- The representative containment leakage for Class 3b sequences is 35La, 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 35La release would not necessarily be consistent with a “Large” release for Peach Bottom Unit 2.
- 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 to 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 2015 population data is adequate for this analysis. Precise evaluations of the projected population 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 1018243 [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 Peach Bottom Unit 2. 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 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 events in 740 reactor years and conservatively assuming a one-year duration for each event. It should be noted that all of the 4 identified large leakage events were PWR events, and the assumption of a one-year duration is not applicable to an inerted containment such as Peach Bottom. NUREG/CR-4220 identifies inerted BWRs as having significantly improved potential for leakage detection because of the requirement to remain inerted during power operation. This calculation presented in NUREG/CR-4220 is called an "upper bound" estimate for BWRs (presumably meaning "inerted" BWR containment designs).

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 is realized from extending the test intervals. For the BWR, the benefit from extending the ILRT frequency from 3 per 10 years to 1 per 10 years was calculated to be a reduction of approximately 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

- Reduced opportunity for draindown events

- Reduced time spent in configurations with impaired mitigating systems

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS draindown events caused by ILRT/LLRT activities, and the other 5 were events involving loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the improvement in the safety due to extending the ILRT test interval.

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., Technical Specification leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Peach Bottom. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551.

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 (and associated NRC SER) [6,7] and Crystal River [20].

The approach included in this guidance document is used in the Peach Bottom Unit 2 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 Liner Corrosion Analysis [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.

EPRI 1018243 [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 Jeffrey'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 Peach Bottom Unit 2 analysis presented here in Section 6.2.

4.2 PLANT-SPECIFIC INPUTS

The Peach Bottom Unit 2 specific information used to perform this ILRT interval extension risk assessment includes the following:

- Level 1 Model results [16]
- Level 2 Model results [16]
- Population within a 50-mile radius [17]

Peach Bottom Unit 2 Internal Events Level 1 PRA Model

The current Level 1 PRA model is an event tree / linked fault tree model characteristic of the as-built, as-operated plant. The total internal events core damage frequency (CDF) used in this analysis is 3.91E-06/yr for Unit 2 [16].

Peach Bottom Unit 2 Internal Events Level 2 PRA Model

The Level 2 Model that is used for Peach Bottom Unit 2 was developed to calculate the LERF contribution as well as the other release categories evaluated in the model. Table 4.2-1 summarizes the pertinent Peach Bottom Unit 2 results in terms of release category (also refer to Table 2-3). The total Large Early Release Frequency (LERF) which corresponds to the H/E release category in Table 4.2-1 was found to be 1.69E-7/yr. The total release frequency is 2.59E-06/yr. With a total CDF of 3.91E-06/yr, this corresponds to an "OK" release limited to normal leakage of 1.32E-6/yr [16].

TABLE 4.2-1
PEACH BOTTOM UNIT 2 LEVEL 2 PRA MODEL RELEASE CATEGORIES AND FREQUENCIES

CATEGORY	FREQUENCY/YR
H/E – High Early (LERF)	1.69E-07
M/E – Medium Early	6.85E-07
L/E – Low Early	0.00E+00
LL/E – Low Low Early	0.00E+00
H/I – High Intermediate	5.50E-08
M/I – Medium Intermediate	7.81E-08
L/I – Low Intermediate	3.96E-08
LL/I – Low Low Intermediate	7.36E-09
H/L – High Late	6.21E-08
M/L – Medium Late	9.79E-07
L/L – Low Late	4.79E-07
LL/L – Low Low Late	3.13E-08
Total Release Frequency	**2.59E-06**
Core Damage Frequency	**3.91E-06**

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for the current Peach Bottom Unit 2 Level 2 model results and more recent population estimates. Each accident sequence was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551. The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 10 bins that are relevant to the analysis. Information from the Peach Bottom Unit 2 PRA Containment Event Trees (CETs) was used to classify each of the Level 2 sequences using these attributes. The definitions of the 10 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 4.2-2 for references purposes. Table 4.2-3 summarizes the calculated population dose associated with each APB from NUREG/CR-4551.

TABLE 4.2-2
COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTIONS [9]

COLLAPSED APB NUMBER	DESCRIPTION
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).

TABLE 4.2-2
COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTIONS [9]

COLLAPSED APB NUMBER	DESCRIPTION
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

TABLE 4.2-3
CALCULATION OF PBAPS POPULATION DOSE RISK AT 50 MILES

COLLAPSED BIN #	FRACTIONAL APB CONTRIBUTIONS TO RISK (MFCR) [1]	NUREG/CR-4551 POPULATION DOSE RISK AT 50 MILES (FROM A TOTAL OF 7.9 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	0.021	0.1659	9.55E-08	1.74E+06
2	0.0066	0.05214	4.77E-08	1.09E+06
3	0.556	4.3924	1.48E-06	2.97E+06
4	0.226	1.7854	7.94E-07	2.25E+06
5	0.0022	0.01738	1.30E-08	1.34E+06
6	0.059	0.4661	2.04E-07	2.28E+06
7	0.118	0.9322	4.77E-07	1.95E+06
8	0.0005	0.00395	7.99E-07	4.94E+03
9	0.01	0.079	3.86E-07	2.05E+05
10	0	0	4.34E-08	0
Totals	**1.0**	**7.9**	**4.34E-6**	

(1) Mean Fractional Contribution to Risk from Table 5.2-3 of NUREG/CR-4551

(2) The total population dose risk at 50 miles from internal events in person-rem is provided 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 2.5-6. 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.

Population Estimate Methodology

The person-rem results in Table 4.2-3 can be used as an approximation of the dose for Peach Bottom. The total population within a 50-mile radius of Peach Bottom is projected to be 5.82E+06 by the year 2015 [17] based on calculating the 10 year growth factor for radial intervals using 1990 and 2000 census data from the SECPOP2000 code. The growth factor was then applied to project out to year 2015. This value is greater than the value of 5.72E+06 based on the Peach Bottom UFSAR [24] for the year 2015 which predicts a 20% increase in the population each decade from 1980 to 2000. The use of the 2015 estimate based on the more recent SECPOP2000 data is judged to be sufficient to perform this assessment.

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 obtain dose estimates for Peach Bottom.

Total Peach Bottom 2015 Population50miles = 5.82E+0 6
Peach Bottom 1980 Population from NUREG/CR-4551 = 3.02E+06
Population Dose Factor = 5.82E+06 / 3.02E+06 = 1.93

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 Peach Bottom 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 Peach Bottom.

TABLE 4.2-4
CALCULATION OF PEACH BOTTOM POPULATION DOSE RISK AT 50 MILES

ACCIDENT PROGRESSION BIN #	NUREG/CR-4551 POPULATION DOSE AT 50 MILES (PERSON-REM)	BIN MULTIPLIER USED TO OBTAIN PEACH BOTTOM POPULATION DOSE	PEACH BOTTOM ADJUSTED POPULATION DOSE AT 50 MILES (PERSON-REM)
1	1.74E+06	1.93	3.36E+06
2	1.09E+06	1.93	2.10E+06
3	2.97E+06	1.93	5.73E+06
4	2.25E+06	1.93	4.34E+06
5	1.34E+06	1.93	2.59E+06
6	2.28E+06	1.93	4.40E+06
7	1.95E+06	1.93	3.76E+06
8	4.94E+03	1.93	9.53E+03
9	2.05E+05	1.93	3.96E+05
10	0	1.93	0.00E+00

Application of Peach Bottom Unit 2 PRA Model Results to NUREG/CR-4551 Level 3 Output

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the current Peach Bottom Unit 2 PRA 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 apply the Peach Bottom Unit 2 PRA Level 2 model results into a format which allowed for the scaling of the Level 3 results based on current Level 2 output. This subsection provides a description of the process used to apply the Peach Bottom Unit 2 PRA Level 2 model results into a form that can be used to generate Level 3 results using the NUREG/CR-4551 documentation. Note that this is the same approach that was used in 2001 ILRT extension for Peach Bottom Unit 3 [23].

The basic process that was pursued to obtain Level 3 results based on the Peach Bottom Unit 2 Level 2 model and NUREG/CR-4551 was to define a useful relationship

between the Level 2 and Level 3 results. Consequently, each non-zero sequence of the Peach Bottom Unit 2 Level 2 model was reviewed and assigned into one of the collapsed Accident Progression Bins (APBs) from NUREG/CR-4551. The Level 2 model contains a significantly larger amount of information about the accident sequences than what is used in the collapsed APBs in NUREG/CR-4551 and this assignment process required simplification of accident progression information and assumptions related to categorizations of certain items. The relevant assumptions used for these assignments are shown in Table 4.2-5. Other nodes are included in the Peach Bottom Unit 2 Level 2 model, but these were not utilized (or did not contribute) to the APB assignment performed here for the ILRT assessment.

TABLE 4.2-5
NODAL ASSUMPTIONS

ACCIDENT CLASS	PBAPS PSA CONTAINMENT EVENT TREE NODE	ASSUMPTION
1	IS – Containment Isolation	If the containment is not isolated, it is assumed that it will be open for the equivalent of an un-scrubbed release as soon as the vessel is breached. No depressurization is asked prior to this node; it is assumed that RPV pressure is >= 200 psi for these sequences. This is bin #3.
	OP – Operator depressurizes the RPV	It is assumed that success on this branch results in RPV pressure below 200 psi.
	RX – Core Melt Arrested in Vessel	A success on this branch signifies that there is no vessel breach. The sequences following this path are grouped in bin #9. However, there is one case in which combustible gas venting (GV) fails followed by containment failure (CZ); this is assumed to result in a high early release and is categorized as a bin #4 event for low pressure and #3 for high pressure.
	CX – Containment Intact During Flood, RPV Breach	Failure of containment during flood is assumed to result in an un-scrubbed release. The timing is technically later than vessel breach, but it is conservatively assumed to be "early" and is grouped in bins 3 or 4 depending on RPV pressure.

TABLE 4.2-5
NODAL ASSUMPTIONS

ACCIDENT CLASS	PBAPS PSA CONTAINMENT EVENT TREE NODE	ASSUMPTION
	NC – No Large Containment Failure	A large containment failure instigated by high containment pressure following vessel breach is assigned to the "late containment failure" bins. The sequences contributing to these bins need to be separated into either WW or DW failures. While the PB CETs distinguish between these types of failures, the NUREG/CR-4551 analysis appears to take credit for scrubbing for any WW release (with respect to the collapsed bins in Section 2.4.3). Not all WW failure in the CETs can be credited with successful scrubbing. Given a large containment failure, the only successful scrubbing path is that in which the WW fails in an area above the water level (success in node WW).
	MU – Coolant Inventory Makeup	Coolant inventory makeup is assumed only to provide cooling to the core debris. No credit is taken for any potential scrubbing effects that water coverage may yield.
	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub.
2	RX – Core Melt Arrested in Vessel	A success on this branch signifies that there is no vessel breach. The sequences following this path are grouped in bin #9.
	CZ/SI – Containment Intact/Mark I Shell Failure	Given that the core melt has not been contained in the RPV, failure in node CZ is assumed to result in an un-scrubbed release through the drywell. Failure in node SI is also assumed to result in an un-scrubbed release due to fission product release through the gap between the liner and the concrete. No credit is given to reactor building scrubbing (RB) or to injection to the DW or RPV (TD). The sequences with failures in these nodes are assigned to bins 3 or 4 depending on RPV pressure.
	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub.
	SP – Suppression Pool Not Bypassed	The suppression pool bypass node is considered in the PB CETs to determine whether the vent volume passes through the suppression pool or not. This node is currently only quantified for cases in which the core melt has been arrested in the RPV (no VB breach). These sequences are assigned to bin #9 and no further breakdown of the sequences is performed.

TABLE 4.2-5
NODAL ASSUMPTIONS

ACCIDENT CLASS	PBAPS PSA CONTAINMENT EVENT TREE NODE	ASSUMPTION
3	MU – Coolant Inventory Makeup	Coolant inventory makeup is assumed only to provide cooling to the core debris. No credit is taken for any potential scrubbing effects that water coverage may yield.
	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub.
	SP – Suppression Pool Not Bypassed	The suppression pool bypass node is considered in the PB CETs to determine whether the vent volume passes through the suppression pool or not. This node is quantified in Class 3 accidents for both vessel breach and "no breach" cases. For no vessel breach: Bin #9 is assigned unless there is a failure in the CZ node. A failure in the CZ node denotes early containment failure and these sequences are assigned to bin #4 (depressurization is always successful in the Class 3 trees, so there is no use of bin #3.) For vessel breach: If the WW is not bypassed, bin #7 is assigned, which is in accord with the bin definition of "vessel breach, vent". If the WW is bypassed, the conditions are assumed to be similar to bin #6 as the venting will take place late in time as would a late containment failure and the un-scrubbed vent volume will be vented directly to the atmosphere through the stack.
	CZ/SI – Containment Intact/Mark I Shell Failure	Given that the core melt has not been contained in the RPV, failure in node CZ is assumed to result in an un-scrubbed release through the drywell. Failure in node SI is also assumed to result in an un-scrubbed release due to fission product release through the gap between the liner and the concrete. No credit is given to reactor building scrubbing (RB) or to injection to the DW or RPV (TD). The sequences with failures in these nodes are assigned to bins 3 or 4 depending on RPV pressure.
4	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub.
	SP – Suppression Pool Not Bypassed	The suppression pool bypass node is considered in the PB CETs to determine whether the vent volume passes through the suppression pool or not. This node is quantified in Class 4 accidents for only "no breach" cases. For no vessel breach Bin #9 is assigned.

TABLE 4.2-5
NODAL ASSUMPTIONS

ACCIDENT CLASS	PBAPS PSA CONTAINMENT EVENT TREE NODE	ASSUMPTION
	CZ/SI – Containment Intact/Mark I Shell Failure	Given that the core melt has not been contained in the RPV, failure in node CZ is assumed to result in an un-scrubbed release through the drywell. Failure in node SI is also assumed to result in an un-scrubbed release due to fission product release through the gap between the liner and the concrete. No credit is given to reactor building scrubbing (RB) or to injection to the DW or RPV (TD). The sequences with failures in these nodes are assigned to bins 3 or 4 depending on RPV pressure.
5	N/A	No collapsed bin is available for containment bypass scenarios. The closest match to a bypass scenario is assumed to be a vessel breach with early drywell failure (bins 3 and 4). These bins are assigned based on RPV pressure (failure to depressurize is set to 0.0, so all sequences with non-zero results will be assigned to bin #4).

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 Peach Bottom Unit 2, as detailed in Section 5, means that 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 Peach Bottom Unit 2 Level 2 PRA 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.

Peach Bottom Unit 2 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 La) and consideration of the performance factors in NEI 94-01, Section 11.3.

Based on completion of two successful ILRTs at Peach Bottom Unit 2, the current ILRT interval is once per ten years. 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 Peach Bottom Unit 2 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 for Peach Bottom Unit 2, 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 as 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.

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. It should be noted that the Calvert Cliffs analysis was performed for a concrete cylinder and dome containment with a steel liner whereas the Peach Bottom containment is a BWR Mark I containment with a steel shell in the drywell region including the portion below the concrete drywell floor. As such, not all aspects of the Calvert Cliffs analysis are directly applicable to Peach Bottom. Each of the analysis steps is described below with their relationship to the Calvert analysis

noted where applicable. The Peach Bottom primary containment is a pressure-suppression BWR/Mark I containment type that also includes a steel-lined reinforced concrete structure.

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 floor and the containment walls
- 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

Assumptions

- Based on a review of industry events, an Oyster Creek incident is assumed to be applicable to Peach Bottom for a concealed shell failure in the floor. In the Calvert Cliffs analysis, this event was assumed not to be applicable and 0.5 failures were assumed (i.e. a typical PRA model when no failures have been identified) (See Table 4.4-1, Step 1.)
- The two corrosion events used to estimate the containment wall flaw probability in the Calvert Cliffs analysis are assumed to be applicable to the Peach Bottom Unit 2 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.
- For consistency with the Calvert Cliffs analysis, the estimated historical flaw probability is limited to 5.5 years 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 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 containment walls and dome region and 0.11% (10% less) for the basemat. These values were determined from an assessment of the containment fragility curve versus the ILRT test pressure. For Peach Bottom Unit 2 the containment failure probabilities are conservatively assumed to be 10% for the drywell outer walls and 1% for the basemat. 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 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 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 WALL		CONTAINMENT BASEMAT	
1	**Historical Steel Liner Flaw Likelihood** Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).	Events: 2 2/(70 * 5.5) = **5.2E-3**		Events: 1 1.0/(70 * 5.5) = **2.6E-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 1 avg 5-10 15	Failure Rate 2.1E-3 5.2E-3 1.4E-2	Year 1 avg 5-10 15	Failure Rate 1.0E-3 2.6E-3 7.0E-3
		15 year average = 6.27E-3		**15 year average = 3.14E-3**	
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 the values are calculated based on the 3, 10, and 15 year intervals.)		**0.36% (1 to 3 years)** **2.03% (1 to 10 years)** **4.70% (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, twice that value is utilized (since 1 failure is assumed applicable instead of 0.5) and the values are calculated based on the 3, 10, and 15 year intervals.)	

TABLE 4.4-1
STEEL LINER CORROSION BASE CASE

STEP	DESCRIPTION	CONTAINMENT WALL	CONTAINMENT BASEMAT
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the containment is assumed to be 10% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 1%, (compared to 0.11% in the Calvert Cliffs analysis).	**10%**	**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.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	0.0071% (at 3 years) 0.71% * 10% * 10% 0.0406% (at 10 years) 4.06% * 10% * 10% 0.0940% (at 15 years) 9.40% * 10% * 10%	0.0036% (at 3 years) 0.36% * 1% * 100% 0.0203% (at 10 years) 2.03% * 1% * 100% 0.0470% (at 15 years) 4.70% * 1% * 100%

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

At 3 years: 0.0071% + 0.0036% = 0.0107%

At 10 years: 0.0406% + 0.0203% = 0.0609%

At 15 years: 0.0940% + 0.0470% = 0.1410%

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 5.0-1 lists these accident classes.

The analysis performed examined Peach Bottom Unit 2-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 5.0-1
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 5.0-1.

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

This step involves the review of the Peach Bottom Unit 2 containment event trees (CETs) and Level 2 accident sequence frequency results. The CETs characterize the response of the containment to important severe accident sequences. As described in Section 4.2, the Peach Bottom Unit 2 CETs were examined and each endstate was applied to one of the Accident Progression Bins as defined in NUREG/CR-4551. The correlation between the NUREG/CR-4551 Accident Progression Bins to the EPRI containment release categories is shown in Table 5.1-1. This application combined with the Peach Bottom Unit 2 dose (person-rem) results determined from Table 4.2-4 forms the basis for estimating the population dose for Peach Bottom.

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). 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 5.0-1 were developed for Peach Bottom Unit 2 based on the assumptions shown in Table 4.2-5, 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 eight containment release class frequencies were developed consistent with the definitions in Table 5.0-1 as described following Table 5.1-1.

TABLE 5.1-1
CONTAINMENT RELEASE TYPE ASSIGNMENT FROM THE NUREG/CR-4551 CONSEQUENCE MODEL

EPRI TR-104285 CONTAINMENT RELEASE		NUREG/CR-4551	
SCENARIO TYPE	DOSE (PERSON-REM)	ACCIDENT PROGRESSION BIN	PEACH BOTTOM UNIT 2 DOSE (PERSON-REM)
1	9.53E+03 (1)	8 (VB, No CF, No Vent)	9.53E+03
		10 (No core damage)	0.00E+00
2 (1)	5.73E+06	3 (VB, Early DW, Hi Press)	5.73E+06
7	3.19E+06(2)	1 (VB, Early WW, Hi Press)	3.36E+06
		2 (VB, Early WW, Lo Press)	2.10E+06
		4 (VB, Early DW, Lo Press)	4.34E+06
		5 (VB, Late WW)	2.59E+06
		6 (VB, Late DW)	4.40E+06
		7 (VB, No CF, Vent)	3.76E+06
		9 (No VB, No CF, No Vent)	3.96E+05
8 (1)	4.34E+06	4 (ISLOCA)	4.34E+06

1. No specific Release Bin for this category exists in NUREG/CR-4551. For simplicity, all sequences assigned to APB #3 is used in this analysis to represent EPRI Class 2 and all LOCAs outside containment sequences assigned to APB #4 are assigned to EPRI Class 8. This will not impact the calculated change for the proposed ILRT extension.

2. Given that multiple NUREG/CR-4551 discrete scenarios apply to the broader EPRI type, the EPRI type dose is based on a weighted average (weights based on Peach Bottom Unit 2 PRA scenario frequencies) of the applicable NUREG/CR-4551 APB doses.

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 for these sequences is 1.22E-06/yr and is determined by subtracting all containment failure end states including the EPRI/NEI Class 3a and 3b frequency calculated below, from the total CDF. For this analysis, the associated maximum containment leakage for this group is 1La, consistent with an intact containment evaluation.

Class 2 Sequences

This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. For simplicity, the frequency is obtained from all sequences that were assigned to APB #3 for Peach Bottom Unit 2 which is 9.47E-08/yr.

Class 3 Sequences

This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small (2La to 35La) or large (>35La). In this analysis, a value of 10La was used for small pre-existing flaws and 35La for relatively large flaws.

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).

Class_3a = 0.027 * (CDF-Class 2-Class 8)
= 0.027 * (3.91E-06 – 9.47E-08 – 9.48E-08) = 1.00E-07/yr

Class_3b = 0.0027 * (CDF-Class 2-Class 8)
= 0.0027 * (3.91E-06 – 9.47E-08 – 9.48E-08) = 1.00E-08/yr

For this analysis, the associated containment leakage for Class 3a is 10La and for Class 3b is 35La. 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. For this analysis, the associated radionuclide releases are based on the application of the Level 2 endstates to the Accident Progression Bins from NUREG/CR-4551 as described in Section 4.2. The Class 7 Sequences are divided into 7 categories which consists of Bins 1, 2, 4 (non-ISLOCA cases), 5, 6, 7, and 9 from NUREG/CR-4551. The failure frequency and population dose for each specific APB is shown below in Table 5.1-2. The total release frequency and total dose are then used to determine a weighted average person-rem for use as the representative EPRI Class 7 dose in the subsequent analysis. Note that the total frequency and dose associated from this EPRI class does not change as part of the ILRT extension request.

TABLE 5.1-2
ACCIDENT CLASS 7 FAILURE FREQUENCIES AND POPULATION DOSES (PEACH BOTTOM UNIT 2 BASE CASE LEVEL 2 MODEL)

ACCIDENT CLASS (APB NUMBER)	RELEASE FREQUENCY/YR	POPULATION DOSE (50 MILES) PERSON-REM (1)	POPULATION DOSE RISK (50 MILES) (PERSON-REM/YR) (2)
7a (APB #1)	0.00E+00	3.36E+06	0.00E+00
7b (APB #2)	0.00E+00	2.10E+06	0.00E+00
7c (APB #4 non-ISLOCA)	1.59E-06	4.34E+06	6.89E+00
7d (APB #5)	1.69E-07	2.59E+06	4.36E-01
7e (APB #6)	9.96E-10	4.40E+06	4.38E-03
7f (APB #7)	1.55E-08	3.76E+06	5.84E-02
7g (APB #9)	6.23E-07	3.96E+05	2.46E-01
Class 7 Total	2.39E-06	3.19E+06(3)	7.63E+00

(1) Population dose values obtained from Table 4.2-4 based on the Accident Progression Bin.

(2) Obtained by multiplying the Release Frequency value from the second column of this table by the Population dose value from the third column of this table.

(3) The weighted average population dose for Class 7 is obtained by dividing the total population dose risk by the total release frequency.

Class 8 Sequences

This group consists of all core damage accident progression bins in which containment bypass occurs. For ISLOCA, the frequency is obtained from all sequences that were assigned to APB #4 for Peach Bottom Unit 2 which is 9.48E-08/yr.

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 definition of Accident Classes defined in EPRI-TR-104285. Table 5.1-3 summarizes these accident frequencies by Accident Class.

TABLE 5.1-3
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (PEACH BOTTOM BASE CASE)

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (PER RX-YR)
1	No Containment Failure	1.22E-06
2(1)	Large Isolation Failures (Failure to Close)	9.47E-08
3a	Small Isolation Failures (liner breach)	1.00E-07
3b	Large Isolation Failures (liner breach)	1.00E-08
4	Small Isolation Failures (Failure to seal –Type B)	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7	Failures Induced by Phenomena (Early and Late)	2.39E-06
8(1)	Bypass (Interfacing System LOCA)	9.48E-08
CDF	**All CET End states (including very low and no release)**	**3.91E-06**

(1) The EPRI Class 2 and Class 8 scenarios are assumed to be LERF in the ILRT methodology, and the sum of these sequence contributions from the simplified APB assignment of 1.90E-07/yr agrees quite well with the Peach Bottom Unit 2 detailed Level 2 PRA model reported LERF value of 1.69E 07/yr.

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 = 9.53E+03 person-rem (at 1.0La)[1]

Class 2 = 5.73E+06 person-rem[2]

Class 3a = 9.53E+03 person-rem x 10La = 9.53E+04 person-rem[3]

Class 3b = 9.53E+03 person-rem x 35La = 3.34E+05 person-rem[3]

Class 4 = Not analyzed

Class 5 = Not analyzed

Class 6 = Not analyzed

Class 7 = 3.19E+06 person-rem[4]

Class 8 = 4.34E+06 person-rem[5]

(1) The Class 1, containment intact sequences, dose is assigned from the APB #8 (No CF, No Vent) from the NUREG/CR-4551 adjusted dose for Peach Bottom Unit 2 as shown in Table 4.2-4.

(2) The Class 2, containment isolation failures, dose is approximated from APB #3 (VB, Early DW, Hi Press) from Table 4.2-4.

(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 the weighted average dose calculated from APBs #1, 2, 5, 6, 7, and 9 from Table 4.2-4 as detailed in Table 5.1-2 above.

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. As an approximation, the releases for this class are assigned from APB #4 (VB, Early DW, Lo Press) from Table 4.2-4.

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
PEACH BOTTOM POPULATION DOSE ESTIMATES FOR POPULATION WITHIN 50 MILES

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	REPRESENTATIVE ACCIDENT PROGRESSION BIN (APB)	DESCRIPTION	PERSON-REM (50 MILES)
1	8	No Containment Failure (1 La)	9.53E+03
2	3	Large Isolation Failures (Failure to Close)	5.73E+06
3a	$10L_a$	Small Isolation Failures (liner breach)	9.53E+04
3b	$35L_a$	Large Isolation Failures (liner breach)	3.34E+05
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	1, 2, 4(non-ISLOCA), 5, 6, 7, 9	Failures Induced by Phenomena (Early and Late)	3.19E+06
8	4 (ISLOCA)	Bypass (Interfacing System LOCA)	4.34E+06

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

TABLE 5.2-2
PEACH BOTTOM 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]	9.53E+03	1.22E-06	1.16E-02	1.22E-06	1.16E-02	-3.78E-06
2	Large Isolation Failures (Failure to Close)	5.73E+06	9.47E-08	5.43E-01	9.47E-08	5.43E-01	--
3a	Small Isolation Failures (liner breach)	9.53E+04	1.00E-07	9.58E-03	1.00E-07	9.58E-03	--
3b	Large Isolation Failures (liner breach)	3.34E+05	1.00E-08	3.35E-03	1.04E-08	3.48E-03	1.32E-04
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.19E+06	2.39E-06	7.63E+00	2.39E-06	7.63E+00	--
8	Bypass (ISLOCA)	4.34E+06	9.48E-08	4.12E-01	9.48E-08	4.12E-01	--
CDF	All CET end states		3.91E-06	8.61E+00	3.91E-06	8.61E+00	1.28E-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.

The calculated dose for Peach Bottom Unit 2 compares favorably with other locations given the relative population densities surrounding each location.

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 Peach Bottom 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 5.3-2.

TABLE 5.3-1
PEACH BOTTOM 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]	9.53E+03	9.58E-07	9.13E-03	9.56E-07	9.11E-03	-2.16E-05
2	Large Isolation Failures (Failure to Close)	5.73E+06	9.48E-08	5.43E-01	9.47E-08	5.43E-01	--
3a	Small Isolation Failures (liner breach)	9.53E+04	3.35E-07	3.19E-02	3.35E-07	3.19E-02	--
3b	Large Isolation Failures (liner breach)	3.34E+05	3.35E-08	1.12E-02	3.57E-08	1.19E-02	7.56E-04
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.19E+06	2.39E-06	7.63E+00	2.39E-06	7.63E+00	--
8	Bypass (ISLOCA)	4.34E+06	9.48E-08	4.12E-01	9.48E-08	4.12E-01	--
CDF	**All CET end states**		**3.91E-06**	**8.64E+00**	**3.91E-06**	**8.64E+00**	**7.34E-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
PEACH BOTTOM 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]	9.53E+03	7.74E-07	7.37E-03	7.68E-07	7.32E-03	-5.00E-05
2	Large Isolation Failures (Failure to Close)	5.73E+06	9.47E-08	5.43E-01	9.47E-08	5.43E-01	--
3a	Small Isolation Failures (liner breach)	9.53E+04	5.02E-07	4.79E-02	5.02E-07	4.79E-02	--
3b	Large Isolation Failures (liner breach)	3.34E+05	5.02E-08	1.68E-02	5.55E-08	1.85E-02	1.75E-03
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	3.19E+06	2.39E-06	7.63E+00	2.39E-06	7.63E+00	--
8	Bypass (ISLOCA)	4.34E+06	9.48E-08	4.12E-01	9.48E-08	4.12E-01	--
CDF	**All CET end states**		**3.91E-06**	**8.66E+00**	**3.91E-06**	**8.66E+00**	**1.70E-03**

(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

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 1E-6/yr and increases in LERF below 1E-7/yr, and small changes in LERF as below 1E-6/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Peach Bottom Unit 2, 100% of the frequency of Class 3b sequences can be used as a 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 Table 5.2-2, the Class 3b frequency is 1.04E-08/yr, which includes the corrosion effect of the containment liner. Based on a ten-year test interval from Table 5.3-1, the Class 3b frequency is 3.57E-08/yr; and, based on a fifteen-year test interval from Table 5.3-2, it is 5.55E-08/yr. 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 (including corrosion effects) is 4.51E-08/yr. Similarly, the increase due to increasing the interval from 10 to 15 years (including corrosion effects) is 1.98E-08/yr. 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 very small change in risk when comparing the 15 year results to the current 10-year requirement, and even 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.

CCFP 3 IN 10 YRS	CCFP 1 IN 10 YRS	CCFP 1 IN 15 YRS	$\Delta CCFP_{15-3}$	$\Delta CCFP_{15-10}$
66.35%	67.00%	67.51%	1.15%	0.51%

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

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

5.6 SUMMARY OF INTERNAL EVENTS RESULTS

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

TABLE 5.6-1
PEACH BOTTOM 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	9.53E+03	1.22E-06	1.16E-02	9.56E-07	9.11E-03	7.68E-07	7.32E-03
2	5.73E+06	9.47E-08	5.43E-01	9.47E-08	5.43E-01	9.47E-08	5.43E-01
3a	9.53E+04	1.00E-07	9.58E-03	3.35E-07	3.19E-02	5.02E-07	4.79E-02
3b	3.34E+05	1.04E-08	3.48E-03	3.57E-08	1.19E-02	5.55E-08	1.85E-02
7	3.19E+06	2.39E-06	7.63E+00	2.39E-06	7.63E+00	2.39E-06	7.63E+00
8	4.34E+06	9.48E-08	4.12E-01	9.48E-08	4.12E-01	9.48E-08	4.12E-01
Total		3.91E-06	8.61E+00	3.91E-06	8.64E+00	3.91E-06	8.66+00

TABLE 5.6-1
PEACH BOTTOM 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
ILRT Dose Rate from 3a and 3b		1.31E-02		4.38E-02		6.64E-02	
Delta Total Dose Rate[1]	From 3 yr	---		2.83E-02		4.91E-02	
	From 10 yr	---		---		2.08E-02	
3b Frequency (LERF)		1.04E-08		3.57E-08		5.55E-08	
Delta LERF	From 3 yr	---		2.53E-08		4.51E-08	
	From 10 yr	---		---		1.98E-08	
CCFP %		66.35%		67.00%		67.51%	
Delta CCFP %	From 3 yr	---		0.65%		1.16%	
	From 10 yr	---		---		0.51%	

1. The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the IRLT frequency.

5.7 EXTERNAL EVENTS CONTRIBUTION

Since the risk acceptance guidelines in RG-1.174 are intended for comparison with a full-scope assessment of risk, including internal and external events, a bounding analysis of the potential impact from external events is presented here.

The results of the PBAPS IPEEE did not provide comprehensive CDF and LERF evaluations. As such, there are no specific CDF and LERF values available from the IPEEE to support the ILRT risk assessment.

Fire Induced Contribution

Since the performance of the IPEEE, a PB Fire Risk Analysis was completed in 2007 [18]. The EPRI FIVE Methodology [27] and Fire PRA Implementation Guide (FPRAIG) [28] screening approaches, EPRI Fire Events Database [29] and plant specific data were used to develop the PB Fire PRA [18]. Based on the 2007 PB Fire PRA update, the PB CDF contribution due to internal fires in the unscreened fire areas is calculated at 4.38E-5/yr for Unit 2. The fire PRA does not quantify the LERF risk measure, however, review of NUREG 1742 [25], indicates that the fire CDF for BWRs is primarily determined by plant transient type of events such that the LERF distribution from the fire CDF can be assumed to be similar to that from the internal events model.

The reported fire PRA CDF values are approximately a factor of 11.2 higher than the internal events CDF values. The fire CDF values are judged to be very conservative given the methods employed in developing the fire PRA for Peach Bottom when compared to the best estimate CDF and LERF values obtained from the internal events models. Given this, it is reasonable to assume that the total impact from external events risk is bounded by assuming a factor of 11.2 on the internal events evaluation. This assumption is used to provide insight into the impact of the total external hazard risk on the conclusions of this ILRT risk assessment.

Using the relationship described in the LAR submittal for PB for the impact on 3b frequency due to increases in the ILRT surveillance interval, the EPRI Category 3b frequency for the 3-per-10 year, 1-per-10 year, and 1-per-15 year ILRT intervals are shown in Table 5.6-1 of the PRA analysis portion of the submittal as 1.04E-08/yr, 3.57E-08/yr, and 5.55E-08/yr, respectively. Therefore, the change in the LERF risk measure due to extending the ILRT from 3-per-10 years to 1-per-15 years, including both internal and external hazard risk, is estimated as shown in Table 5.7-1 below:

TABLE 5.7-1
PB2 3B (LERF) AS A FUNCTION OF ILRT FREQUENCY FOR INTERNAL AND EXTERNAL EVENTS (INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

	3B FREQUENCY (3-PER-10 YR ILRT)	3B FREQUENCY (1-PER-10 YEAR ILRT)	3B FREQUENCY (1-PER-15 YEAR ILRT)	LERF INCREASE[1]
Internal Events Contribution	1.04E-08	3.57E-08	5.55E-08	4.51E-8/yr
External Events Contribution	1.16E-07	4.00E-07	6.22E-07	5.05E-7/yr
Combined (Internal + External)	1.27E-07	4.36E-07	6.77E-07	5.50E-7/yr

(1) Associated with the change from the current 3-per-10 year frequency to the proposed 1-per-15 year frequency

Thus, the increase in LERF (ΔLERF) due to the combined internal and external events contribution is estimated as 5.50E-7/yr.

NRC Regulatory Guide 1.174 [32], "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant. As discussed in Section 2 of this PRA analysis, the risk acceptance criteria of RG 1.174 is used here to assess the ILRT interval extension.

The 5.50E-7/yr increase in LERF due to the combined internal and external events from extending the Peach Bottom ILRT frequency from 3-per-10 years to 1-per-15 years falls into Region II between 1E 7 to 1E-6 per reactor year ("Small Change" in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the "Small Change" range, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr. Similar bounding assumptions regarding the external event contributions that were made above (i.e. a factor of 11.2 compared to the internal events results) is used for the total LERF estimate.

From Table 4.2-1, the LERF due to internal event accidents is 1.7E-7/yr. With this information, the bounding LERF impact due to external events is estimated based on the discussion above.

TABLE 5.7-2
IMPACT OF 15-YR ILRT EXTENSION ON LERF (3B)

Internal Events LERF	1.7E-7/yr
External Events LERF	1.9E-6/yr
Internal Events LERF due to ILRT (at 15 years)[1]	5.55E-8/yr
External Events LERF due to ILRT (at 15 years)[1]	6.22E-7/yr
Total:	2.75E-6/yr

[1] Including age adjusted steel liner corrosion likelihood.

As can be seen, the estimated upper bound LERF for Peach Bottom is estimated at 2.75E-6/yr, which is less than the RG 1.174 requirement to demonstrate that the total LERF of internal events and external events is less than 1E-5/yr.

6.0 SENSITIVITIES

6.1 SENSITIVITY TO CORROSION IMPACT ASSUMPTIONS

The results in Tables 5.2-2, 5.3-1, and 5.3-2 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. Additionally, it should be noted that the Torus Corrosion Monitoring Program at Peach Bottom should be a more effective method than Appendix J Type A tests (ILRT) for identifying degrading minimum wall conditions, since the Type A test will only identify an actual breach in the pressure boundary.

In any event, 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 containment wall and 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 1.48E-07/yr. 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)	INCREASE IN CLASS 3B FREQUENCY (LERF) FOR ILRT EXTENSION 3 TO 15 YEARS (PER RX-YR)	
			TOTAL INCREASE	INCREASE DUE TO CORROSION
Base Case Doubles every 5 yrs	Base Case (10% Wall, 1.0% Basemat)	Base Case 10% Wall, 100% Basemat	4.50E-08	4.85E-09
Doubles every 2 yrs	Base	Base	5.13E-08	1.11E-08
Doubles every 10 yrs	Base	Base	4.43E-08	4.09E-09
Base	Base	15% Wall	4.66E-08	6.47E-09
Base	Base	5% Wall	4.34E-08	3.23E-09
Base	100% Wall, 10% Basemat	Base	8.87E-08	4.85E-08
Base	1.0% Wall, 0.1% Basemat	Base	4.07E-08	4.85E-10
LOWER BOUND				
Doubles every 10 yrs	1.0% Wall, 0.1% Basemat	5% Wall 100% Basemat	4.05E-08	2.73E-10
UPPER BOUND				
Doubles every 2 yrs	100% Wall, 10% Basemat	15% Wall 100% Basemat	1.88E-07	1.48E-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 leaks 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 a 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 Jeffrey'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 La for small and 35 La for large) are used here. Table 6.2-1 illustrates the magnitudes and probabilities of a pre-existing leak in containment associated with the Jeffrey'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, and 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)	JEFFREY'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. As mentioned previously, probability

values are those associated with the magnitude of the leakage used in the Jeffrey's non-informative prior evaluation (10La for small and 35La 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 La 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 1.47E-08/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is 6.13E-09/yr. As such, if the expert elicitation mean probability of occurrences are used instead of the non-informative prior estimates, the change in LERF for Peach Bottom Unit 2 is even further below the threshold criteria for a "very small" change in risk when compared to the current 1-in-10 or 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
PEACH BOTTOM 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	9.53E+03	1.31E-06	1.25E-02	1.27E-06	1.21E-02	1.24E-06	1.18E-02
2	5.73E+06	9.47E-08	5.43E-01	9.47E-08	5.43E-01	9.47E-08	5.43E-01
3a	9.53E+04	1.44E-08	1.38E-03	4.81E-08	4.58E-03	7.22E-08	6.88E-03
3b	3.34E+05	3.67E-09	1.22E-03	1.22E-08	4.08E-03	1.83E-08	6.12E-03
7	3.19E+06	2.39E-06	7.63E+00	2.39E-06	7.63E+00	2.39E-06	7.63E+00

TABLE 6.2-2
PEACH BOTTOM 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
8	4.34E+06	9.48E-08	4.12E-01	9.48E-08	4.12E-01	9.48E-08	4.12E-01
Total		3.91E-06	8.60E+00	3.91E-06	8.61E+00	3.91E-06	8.61E+00
ILRT Dose Rate from 3a and 3b		2.60E-03		8.66E-03		1.30E-02	
Delta Total Dose Rate[1]	From 3 yr	---		5.66E-03		9.71E-03	
	From 10 yr	---		---		4.05E-03	
3b Frequency (LERF)		3.67E-09		1.22E-08		1.83E-08	
Delta LERF	From 3 yr	---		8.55E-09		1.47E-08	
	From 10 yr	---		---		6.13E-09	
CCFP %		66.18%		66.40%		66.56%	
Delta CCFP %	From 3 yr	---		0.22%		0.38%	
	From 10 yr	---		---		0.16%	

1. The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the IRLT frequency.

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 estimated as 4.02E-08/yr using the NEI guidance as written, and at 1.47E-08/yr using the EPRI Expert Elicitation methodology. 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 for the base case with corrosion included is 4.51E-08. In both cases, the NEI guidance and the EPRI Expert Elicitation Methodology, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.
- 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 4.75E-02 person-rem/yr using the NEI guidance, and drops to 9.71E-03 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 1.03% using the NEI guidance, and drops to about 0.38% using the EPRI Expert Elicitation methodology. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.
- Since the increase in LERF falls well below the "small change" category using the acceptance guidelines of Reg. Guide 1.174, a detailed examination of the external events impact is not explicitly required, nor is it expected to change the conclusions from this assessment.
- To confirm the expected impact from external events, an additional bounding assessment of the potential impact from the risk associated with external events was done. As shown in Table 5.7-1, the total increase in LERF due to internal events and the bounding external

events assessment is 5.50E-07/yr, which is in Region II of the Reg. Guide 1.174 acceptance guidelines.

- Finally, the same bounding analysis indicates that the total LERF from internal and external risks as shown in Table 5.7-2 is 2.75E-06/yr, which is less than the Reg. Guide 1.174 limit of 1E-05/yr given that the ΔLERF is in Region II.

Therefore, increasing the ILRT interval to 15 years is considered to be insignificant since it represents a very small change to the Peach Bottom Unit 2 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 Peach Bottom Unit 2 confirm these general findings on a plant specific basis considering the severe accidents evaluated for Peach Bottom Unit 2, the Peach Bottom containment failure modes, and the local population surrounding Peach Bottom.

8.0 REFERENCES

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APPENDIX A

PRA Technical Adequacy

A.1 Overview

A technical Probabilistic Risk Assessment (PRA) analysis is presented in this calculation to help support a one-time extension of the Peach Bottom Unit 2 containment Type A test integrated leak rate test (ILRT) interval from ten years to fifteen years.

The analysis follows the guidance provided in Regulatory Guide 1.200, Revision 1 [1], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The guidance in RG-1.200 indicates that the following steps should be followed to perform this study:

1. Identify the parts of the PRA used to support the application
   - SSCs, operational characteristics affected by the application and how these are implemented in the PRA model
   - A definition of the acceptance criteria used for the application
2. Identify the scope of risk contributors addressed by the PRA model
   - If not full scope (i.e. internal and external), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.
3. Summarize the risk assessment methodology used to assess the risk of the application
   - Include how the PRA model was modified to appropriately model the risk impact of the change request.
4. Demonstrate the Technical Adequacy of the PRA
   - Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.
   - Document peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.
   - Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the Regulatory Guide (currently, in RG-1.200 Rev. 1 this is just the internal events PRA standard). Provide justification to show that where specific

requirements in the standard are not met, it will not unduly impact the results.

- Identify key assumptions and approximations relevant to the results used in the decision-making process.

Items 1 through 3 are covered in the main body of this calculation. The purpose of this appendix is to address the requirements identified in item 4 above.

A.2 Technical Adequacy of the PRA Model

The PB205C update to the PB PRA model is the most recent evaluation of the risk profile at Peach Bottom for internal event challenges. The PB PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the PB PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

Exelon Generation Company (EGC) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating EGC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Peach Bottom PRA.

PRA Maintenance and Update

The EGC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the EGC Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation procedures. EGC procedure ER-AA-600-1015, "FPIE PRA Model Update" delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating EGC nuclear generation sites. The overall EGC Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model

updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years.

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

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

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant.

EGC will be performing a regularly scheduled update to the Peach Bottom PRA model in 2009, which is expected to be approved in the first half of 2010.

As indicated previously, RG-1.200 also requires that additional information be provided as part of the LAR submittal to demonstrate the technical adequacy of the PRA model used for the risk assessment. Each of these items (plant changes not yet incorporated into the PRA model, relevant peer review findings, consistency with applicable PRA Standards, and the identification of key assumptions) will be discussed in turn.

A.2.1 Plant Changes Not Yet Incorporated into the PRA Model

A PRA updating requirements evaluation (URE-EGC PRA model update tracking database) is created for all issues that are identified that could impact the PRA model. The URE database includes the identification of those plant changes that could impact the PRA model. A review of the current open items in the URE database for Peach Bottom identified a few items with potential impact. These are summarized in the table which follows along with an assessment of the impact for this application.

TABLE A-1
IMPACT OF PLANT CHANGES SINCE THE LAST UPDATE ON THE PEACH BOTTOM PRA MODEL

URE NUMBER	PLANT CHANGE	IMPACT ON THE PBAPS PRA
PB2008-002	A change was made in SE-11 (loss of offsite power procedure) for placement of the Load Tap Changer to be maintained in manual at an optimum tap position of 25 for the entire SBO load sequence.	No impact. A detailed review of the proposed changes to SE-11.1 as documented in the ECR was performed and compared to the human reliability assessment. The changes and movement of the steps has no numerical impact on the Human Error Probability evaluation. However, the next update of the HRA Notebook should include reference to the revised steps that will be included in revision to SE-11.1.
PB2007-024	An Engineering Change Request (ECR) was created for converting the cable spreading room and computer room cardox fire suppression control systems from automatic systems to manual systems.	No impact. Conversion of the cable spreading room and computer room cardox fire suppression control systems from automatic to manual does not impact the PBAPS Fire PRA. Automatic suppression is not credited for fires in the cable spreading room or computer room, which are evaluated in the analysis of Fire Compartment 25.

TABLE A-1
IMPACT OF PLANT CHANGES SINCE THE LAST UPDATE ON THE PEACH BOTTOM PRA MODEL

URE NUMBER	PLANT CHANGE	IMPACT ON THE PBAPS PRA
PB2006-040	An ECR was developed to disable the alarm and system trouble alarm functions associated with the unit 2 main turbine differential expansion detector. No turbine trips will be disabled.	Minimal impact. The disabling of the alarm and system trouble alarm functions should not impact the turbine trip frequency since the trip setpoints are not being modified by this change. In any event, the Turbine Trip Initiating Event Frequency is based on plant-specific data collection efforts, and will be updated as part of the normal PRA model update process.

A.2.2 Applicability of Peer Review Findings and Observations

Several assessments of technical capability have been made, and continue to be planned, for the PBAPS, Units 2 and 3 PRA models. These assessments are as follows and further discussed in the paragraphs below.

- An independent PRA peer review was conducted under the auspices of the BWR Owners Group in 1998, following the Industry PRA Peer Review process [2]. This peer review included an assessment of the PRA model maintenance and update process.
- In 2004, a gap analysis was performed against the available version of the ASME PRA Standard [3] and the draft version of Regulatory Guide 1.200, DG-1122 [4]. In 2006, an assessment of the extent to which the previously defined gaps had been addressed was performed in conjunction with a PRA model update.
- During 2005 and 2006 the PBAPS, Units 2 and 3 PRA model results were evaluated in the BWR Owners Group PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process.
- As part of the PRA model update in 2009, the gap analysis will be updated to reflect pertinent changes to both the PRA Standard and Regulatory Guide 1.200.

A summary of the disposition of 1998 Industry PRA Peer Review facts and observations (F&Os) for the PBAPS, Units 2 and 3 PRA models was documented as part of the statement of PRA capability for MSPI in the PBAPS MSPI Basis Document [5]. As noted in that document, there were no significance level A F&Os from the peer review,

and all significance level B F&Os were addressed and closed out with the completion of the current PB205 and PB305 models of record. Also noted in that submittal was the fact that, after allowing for plant-specific features, there are no MSPI cross-comparison outliers for PBAPS (refer to the third bulleted item above).

A Gap Analysis for the 2002 PBAPS, Units 2 and 3 PRA models (PB202 and PB302, respectively) was completed in January 2004. This Gap Analysis was performed against PRA Standard RA-S-2002 [3] and associated NRC comments in draft regulatory guide DG-1122 [4], the draft version of Regulatory Guide 1.200 Revision 0. This gap analysis defined a list of 83 supporting requirements from the Standard for which potential gaps to Capability Category II of the Standard were identified. For each such potential gap, a PRA updating requirements evaluation (URE-EGC model update tracking database) was documented for resolution.

A PRA model update was completed in 2006, resulting in the PB205C and PB305C updated models. In updating the PRA, changes were made to the PRA to address most of the identified gaps, as well as to address other open UREs. Following the update, an assessment of the status of the gap analysis relative to the new model and the updated requirements in Addendum A of the ASME PRA Standard concluded that 59 of the gaps were fully resolved (i.e., are no longer gaps), and another seven were partially resolved.

All remaining gaps will be reviewed for consideration during the 2009 model update process but are judged to have low impact on the PRA model or its ability to support a full range of PRA applications. The remaining gaps are documented in the URE database so that they can be tracked and their potential impacts accounted for in applications where appropriate.

A.2.3 Consistency with Applicable PRA Standards

As indicated above, A PRA model update was completed in 2006, resulting in the PB205C and PB305C updated models. In updating the PRA, changes were made to the PRA to address most of the identified gaps, as well as to address other open UREs. In 2008, two additional gaps were identified during the performance of a subsequent

review [6] based on additional SRs that were added to Addendum B of the PRA standard [7], the criteria in RG 1.200, Revision 1 including the NRC positions stated in Appendix A of RG 1.200, Revision 1 and further issued clarifications [8, 9]. The results of that review lead to the supporting requirements (SRs) listed below as not meeting Category II in the PRA model used for this assessment. These SRs are summarized in the table which follows along with an assessment of the impact for this application.

Note that for this application, the accepted methodology involves a bounding approach to estimate the change in LERF from extending the ILRT interval. Rather than exercising the PRA model itself, it involves the establishment of separate calculations that are linearly related to the plant CDF contribution that is not already LERF. Consequently, a reasonable representation of the plant CDF that is not LERF does not require that Capability Category II be met in every aspect of the modeling if the Category I treatment is conservative or otherwise does not significantly impact the results.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #1	Update the Peach Bottom ISLOCA evaluation to be consistent with more recent Exelon evaluations.	IE-C12	Open – ISLOCA update has not yet been performed. However, the current ISLOCA values are reasonably conservative compared to other sites that have utilized the more detailed methodology.	Not significant given that the current approach is reasonably conservative.
Gap #2	Interview plant operations, maintenance, engineering, and safety analysis personnel for the purpose of identifying potential IEs that may have been overlooked. Alternatively, have such personnel review Section 2 of the Peach Bottom PRA IE Notebook and provide comments to this effect. Incorporate results of these interviews/reviews as an appendix to the IE Notebook or as a set of appropriate sentences (with references) to Section 2 of the IE Notebook.	IE-A6	Open – Although this would be an enhancement to the IE Notebook, it is not judged as a high priority. The current IE evaluation provides thorough documentation of the Initiating Events considered in the PBAPS model that is consistent with other BWRs. Note that Cat I for this SR does not require the performance of interviews for this purpose.	None. Category I is met and appropriate for this application.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #6	Development of a Peach Bottom PRA Dependency Matrix Notebook such that it becomes the notebook describing the approach to treatment of all the various type of dependencies throughout the PRA should be considered. This can be accomplished by summarizing how all the various aspects of dependencies are treated and where the associated analyses for the dependencies (e.g., supporting walkdown information, room cooling assumptions, water supply duration, HRA, CCF) are documented.	SC-A4	Partially resolved - Although a specific dependency matrix notebook has not been prepared for Peach Bottom, each of the system notebooks includes a description of all dependencies and includes a detailed dependency matrix. Additionally, accident sequence dependencies as a function of initiating event category are discussed in the event tree notebook.	None. Dependencies are modeled. This is simply a baseline PRA model documentation consideration.
Gap #11	Provide descriptions of the limitations of thermal hydraulic analyses with respect to their use in the PRA (bases for success criteria, HRA timing, etc.) and ensure the application is within the limits of the code. Assessments of the capability limitations may be limited to the specific application of the calculation.	SC-C2	Open - Included in practice. MAAP was not utilized outside the bounds of known acceptability. Otherwise, awaiting guidance from EPRI and endorsement from NRC.	None. The model is not used beyond its known limitations. This is a documentation consideration only.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #25	To meet the requirements of SR HR-A1 and HR-B1, the following would be developed as supporting documentation for PBAPS: - A list of the PRA systems to consider for test and maintenance actions - Rules for identifying and screening test and maintenance actions from the PRA - A list of procedures reviewed, the potential test and maintenance actions associated with the procedures, and the disposition of the action (screened or evaluated).	HR-A1 HR-B1	Open – Pre-initiator errors are included for some risk significant systems (i.e. HPCI, RCIC, LPCS, and SLCS) on a generic basis. The performance of a detailed process for identifying and screening test and maintenance pre-initiators is judged to have a minimal impact on the results of the model.	Not significant. Capability Category I is believed to be met for HR-B1. The pre-initiator assessment that exists is adequate for this application. Pre-initiator human actions do not contribute significantly to the risk significance results for this application.
Gap #26	To meet the requirements of SR HR-A2, the following would be developed as supporting documentation for PBAPS: - A list of the PRA systems to consider for mis-calibration actions - Rules for identifying and screening mis-calibration actions from the PRA - A list of procedures reviewed, the potential mis-calibration actions associated with the procedures, and the disposition of the action (screened or evaluated).	HR-A2	Partially Resolved - The process did not include a procedure review but did include a review of the need for transmitter/trip unit components to function properly, or where false signals could prematurely terminate the system function.	Not significant. The pre-initiator assessment that exists is adequate for this application, given the evaluation that has been performed and reflected in the model.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #27	To meet the requirements of SR HR-A3, the following would be developed as supporting documentation for PBAPS: - A list of the PRA systems to consider for common cause mis-calibration actions - Rules for identifying and screening common cause mis-calibration actions from the PRA - A list of procedures reviewed, the potential common cause mis-calibration actions associated with the procedures, and the disposition of the action (screened or evaluated).	HR-A3	Partially Resolved - The process did not include a procedure review but did include a review of the need for transmitter/trip unit components to function properly, and common cause mis-calibrations were also included.	Not significant. The pre-initiator assessment that exists is adequate for this application, given the evaluation that has been performed and reflected in the model.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #31	Establish the 'significant' pre-initiator HFEs based on the DG-1122 definition, and re-quantify the balance of the significant HFEs using the methodologies outlined in PB02AF-003.	HR-D2	Partially Resolved – Pre-initiators were included in the system models as described in the system notebooks that were created as part of the 2005 update. The process included a review of the need for transmitter/trip unit components to function properly, or where false signals could prematurely terminate the system function. However, not all significant pre-initiators were evaluated with a detailed HEP analysis. Rather, they were assigned a 'type' based on the transmitter it is associated with, and the types were assigned an HEP value based on the limited set of detailed pre-initiator evaluations that were performed as described in the HRA notebook.	None. The updated PRA model meets HR-D2 at Capability Category I, which is sufficient for this application.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #54	Document and employ the methodology used for determining the standby component number of demands to include plant specific: a) surveillance tests, b) maintenance acts, c) surveillance tests or maintenance on other components, d) operational demands. Additional demands from post-maintenance testing should not be included.	DA-C6	Open - For the most part, the estimated demands were determined from the Maintenance Rule database, but a confirmation that it is collected exactly consistent with the DA-C6 requirements has not been performed. This is judged to have a minimal impact on the Bayesian updated reliability values utilized in the model.	Not significant. The model is reasonably consistent with data from the plant MR database, which is adequate for this application.
Gap #55	To be consistent with SR DA-C8, the Peach Bottom PRA would need to be enhanced to include reviews of operating experience to determine the times that components were in standby.	DA-C8	Open – Note that Category I allows for estimates of standby status estimates as an acceptable approach.	None. Capability Category I is met, which is adequate for this application.
Gap #58	Ensure that the enhancements associated with DA-C4 include the guidance regarding the definition of maintenance hours that is provided in SR DA-C11 and that the counting of unavailability hours follows that definition.	DA-C11	Open - The maintenance rule data is used directly, but a confirmation that it was collected exactly consistent with the DA-C11 requirements has not been performed. This is judged to have a minimal impact on the unavailability hours used in the model.	Not significant. The model is reasonably consistent with data from the plant MR database, which is adequate for this application.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #59	Ensure that the enhancements associated with DA-C4 include 1) the guidance regarding the treatment of maintenance hours vs. plant operational status that is provided in SR DA-C12 (and ensure that the counting of unavailability hours follows that definition); and 2) perform interviews of maintenance staff for equipment with incomplete or limited maintenance information.	DA-C12	Open - The maintenance rule data is used directly, but a confirmation that it was collected exactly consistent with the DA-C11 requirements has not been performed. This is judged to have a minimal impact on the unavailability hours used in the model.	Not significant. The model is reasonably consistent with data from the plant MR database, which is adequate for this application.
Gap #60	To be consistent with SR DA-C13, the PRA should include an examination of coincident outage times for redundant equipment (both intra- and inter-system) and incorporate the results into the modeling and documentation. However, it is judged that it is not practical to model all potential combinations of coincident maintenance unavailability values, and that a review of maintenance experience would not be sufficient to allow the prediction of the dominant risk contributor combinations. As such, the approach suggested is to identify dominant risk contributor combinations based on knowledge of the accident sequences modeling, and model such combinations of coincident maintenance outages in the fault tree logic. A review of recent maintenance experience would then be performed to identify events of coincident maintenance outages for these combinations to support probability estimation for the events.	DA-C13	Partially resolved - Model includes coincident outage times for a few pertinent combinations (e.g. HPCI/RCIC, RHR Loops), but since no known overlap existed for these combinations, an arbitrarily small value (1.0E-5) was assigned. It is judged that the incorporation of coincident maintenance terms will have a minimal impact on the results of the model.	Not significant. The model is reasonably consistent with known plant operating practice and experience. An exhaustive assessment is not needed to support use of the PRA for this application.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #65	During the plant specific data update, ensure the data used reflects the current design and operating conditions. Include guidance in the documentation related to updating data when changes are made to equipment or operating conditions.	DA-D7	Partially resolved - The Component Data Notebook includes development of the updated plant-specific data evaluation. Specific guidance on updating data when changes are made is not provided, but providing these definitions should not have an impact on the quantitative results from the PRA model.	None. The process used is appropriate. This is a documentation issue only.
Gap #67	The Peach Bottom PRA appropriately includes a number of internal flood initiators and associated event trees (refer to Section 9 of the main documentation). The internal flooding analysis needs to be expanded into a single comprehensive analysis, and updated where appropriate. Flooding documentation needs to be upgraded especially for walkdowns and descriptions of calculations supporting the quantitative analysis.	IF-F*	Partially resolved – Internal Flood analysis updated in 2008. The results of that analysis will be incorporated into the base model as part of the 2009 model update process.	Not significant. The updated internal flooding analysis that has not yet been integrated into the updated model indicates that the contribution from internal flooding initiators to the internal events CDF and LERF risk metrics are still relatively small.
Gap #68	Identify the PRA modeled SSCs in flood areas per requirements of IF-A2 and IF-A3.	IF-A*	Partially resolved – Internal Flood analysis updated in 2008, but not yet integrated into model.	Not significant. See Gap #67.
Gap #69	Identify and document potential flood sources for areas that do not screen out per the requirements in IF-B1, B2, B3, and B4.	IF-B*	Partially resolved – Internal Flood analysis updated in 2008, but not yet integrated into model.	Not significant. See Gap #67.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #70	Identify and document scenarios, propagation paths, and affected SSCs per the requirements in IF-C1, C2, C3, and C4.	IF-C*	Partially resolved – Internal Flood analysis updated in 2008, but not yet integrated into model.	Not significant. See Gap #67.
Gap #71	Identify human actions for flood mitigation and incorporate into model per IF-C6 standards	IF-C6	Partially resolved – Internal Flood analysis updated in 2008, but not yet integrated into model.	Not significant. See Gap #67.
Gap #72	Review and update flood frequencies per IF-D2, D3, D4, and D5.	IF-D*	Partially resolved – Internal Flood analysis updated in 2008, but not yet integrated into model.	Not significant. See Gap #67.
Gap #73	Review and develop flood sequences per the requirements of IF-E1, E2, E3, E4, E5, and E6.	IF-E*	Partially resolved – Internal Flood analysis updated in 2008, but not yet integrated into model.	Not significant. See Gap #67.
Gap #77	The uncertainty analysis could be further enhanced by providing a discussion of the guidelines used to review results and identify important contributors to uncertainty. Use of a systematic process of identifying these areas and evaluating them may improve the overall quality of the analysis.	QU-E2	Partially resolved – Sensitivity studies included as part of the evaluation in Section 4.5 of the PB PRA Summary Notebook, but the choice of sensitivities could be judged as not a systematic process. However, the QU-E2 SR definition has since changed – refer to Add #2 below.	See Add #2.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Gap #80	Include an assessment of the significance of assumptions on the quantitative results.	QU-F4	Open - Identification of key assumptions will be application specific. Also, the QU-F4 SR has been redefined.	See Add #2.
Gap #83	Strict reading of LE-E4 would indicate that the following enhancements to the LERF analysis and associated documentation would need to be made to comply with the Standard: - Explicitly assess dependencies among Level 2 HEPs (and combinations of Level 2 HEPs with Level 1 HEPs) - Perform quantitative sensitivity studies of the LERF analysis- Perform quantitative uncertainty assessment of the LERF analysis.	LE-E4	Open – An update to the Level 2 Analysis is continuing from the effort started in 2008 and will be finalized as part of the 2009 update process.	Minimal impact. The analysis performed in this application is structured around linear relationships to the non-LERF portion of the CDF risk profile. Any increase to the base case LERF values would only reduce the relative changes in LERF calculated for this application.
Add #1	Addendum B of the ASME PRA Standard [7] added SRs to document the quantitative definition used for significant basic event, significant cutset, significant accident sequence, and significant accident progression sequence in the CDF and LERF analysis.	QU-F6 LE-G6	Open – These new SRs will be addressed during the next full PRA model update, but providing these definitions should not have an impact on the quantitative results from the PRA model.	None. This is a documentation issue. The model is not being changed to address this item.

TABLE A-2
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

TITLE	DESCRIPTION OF GAP	APPLICABLE SRS	CURRENT STATUS / COMMENT	IMPORTANCE TO APPLICATION
Add #2	Several SRs associated with treatment of model uncertainty and related model assumptions have been recently redefined. NRC has issued [8] a clarification to its endorsement of the PRA Standard. NRC and EPRI are currently finalizing guidance on an acceptable process for meeting these requirements.	QU-E1 QU-E2 QU-E4 QU-F4 IE-D3 AS-C3 SC-C3 SY-C3 HR-I3 DA-E3 IF-F3 LE-G4	Open – These recently redefined SRs will be addressed during the next full PRA model update.	An initial assessment based on the final EPRI guidance [10] for the base PRA model has been performed. However, as described below in the next section, this does not result in the identification of any key assumptions for this application.

A.2.4 Identification of Key Assumptions

The methodology employed in this risk assessment followed the NEI guidance and utilized the same process that has been utilized in several similar relief requests (including an earlier request for Peach Bottom Unit 3). The analysis included the incorporation of several sensitivity studies and factored in the potential impacts from external events in a bounding fashion. None of the sensitivity studies or bounding analysis indicated any source of uncertainty or modeling assumption that would have resulted in exceeding the acceptance guidelines. Since the accepted process utilizes a bounding analysis approach which is mostly driven by that CDF contribution which does not already lead to LERF, there are no identified key assumptions or sources of uncertainty for this application (i.e. those which would change the conclusions from the risk assessment results presented here).

A.3 Summary

A PRA technical adequacy evaluation was performed consistent with the requirements of RG-1.200, Revision 1. This evaluation combined with the details of the results of this analysis demonstrates with reasonable assurance that the proposed one-time extension to the ILRT interval for Peach Bottom Unit 2 from ten to fifteen years satisfies the risk acceptance guidelines in RG 1.174.

A.4 References

[1] Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities, Revision 1, January 2007.

[2] Boiling Water Reactors Owners' Group, BWROG PSA Peer Review Certification Implementation Guidelines, Revision 3, January 1997.

[3] American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-S-2002, New York, New York, April 2002.

[4] U.S. Nuclear Regulatory Commission, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Draft Regulatory Guide DG-1122, November 2002.

[5] Peach Bottom MSPI Basis Document, Rev. 2, March 27, 2007.

[6] Letter from P. B. Cowan (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, Response to Request for Supplemental Information Associated with Relief Request I4R-44, May 13, 2008 (Adams Accession Number ML081350177).

[7] American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, (ASME RA-S-2002), Addenda RA-Sa-2003, and Addenda RA-Sb-2005, December 2005.

[8] U.S. Nuclear Regulatory Commission Memorandum to Michael T. Lesar from Farouk Eltawila, "Notice of Clarification to Revision 1 of Regulatory Guide 1.200," for publication as a Federal Register Notice, July 27, 2007.

[9] ASME Committee on Nuclear Risk Management in collaboration with ANS Risk Informed Standards Committee, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-S-2008a, For Recirculation Ballot, October 2008.

[10] Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, EPRI, Palo Alto, CA: December 2008 (Final). 1016737.