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NL-14-014

February 4, 2014

U.S. Nuclear Regulatory Commission
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SUBJECT: Proposed License Amendment Regarding Extending the Containment Type A Leak Rate Testing Frequency to 15 years
Indian Point Unit Number 3
Docket No. 50-286
License No. DPR-64

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (Entergy) hereby requests a License Amendment to Operating License DPR-64, Docket No. 50-286 for Indian Point Nuclear Generating Unit No. 3 (IP3). The proposed TS change contained herein would revise Appendix A, Technical Specifications (TS), to allow extension of the ten-year frequency of the Type A or Integrated Leak Rate Test (ILRT) that is required by Technical Specification (TS) 5.5.15 to 15 years on a permanent basis.

Entergy has evaluated the proposed change in accordance with 10 CFR 50.91(a)(1) using the criteria of 10 CFR 50.92(c) and Entergy has determined that this proposed change involves no significant hazards, as described in Attachment 1. The marked up pages showing the proposed changes are provided in Attachment 2. An assessment of the risk impact of extending the ILRT interval is provided in Attachment 3. A copy of this application and the associated attachments are being submitted to the designated New York State official in accordance with 10 CFR 50.91.

Entergy requests approval of the proposed amendment in one calendar year and an allowance of 30 days for implementation. There are no new commitments being made in this submittal. If you have any questions or require additional information, please contact Mr. Robert Walpole, Manager, Regulatory Assurance at (914) 254-6710.

A001
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 4th, 2014.

Sincerely,

JAV/sp

Attachments:
1. Analysis of Proposed Technical Specification Changes Regarding 15 Year Containment ILRT
2. Marked Up Technical Specifications Pages for Proposed Changes Regarding 15 Year Containment ILRT
3. Risk Impact of Extending the ILRT interval Associated with the Proposed Technical Specification Changes

cc: Mr. Douglas Pickett, Senior Project Manager, NRC NRR DORL
Mr. William Dean, Regional Administrator, NRC Region 1
NRC Resident Inspector
Mr. Francis J. Murray, Jr., President and CEO, NYSERDA
Ms. Bridget Frymire, New York State Dept. of Public Service

ATTACHMENT 1 TO NL-14-014

ANALYSIS OF PROPOSED TECHNICAL SPECIFICATION CHANGES

REGARDING 15 YEAR CONTAINMENT ILRT

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

1.0 DESCRIPTION

Entergy Nuclear Operations, Inc. (Entergy) is requesting an amendment to Operating License DPR-64, Docket No. 50-286 for Indian Point Nuclear Generating Unit No. 3 (IP3). The proposed Technical Specification (TS) change contained herein would revise Appendix A, TS, to allow extension of the ten-year frequency of the Type A or Integrated Leak Rate Test (ILRT) that is required by TS 5.5.15 to 15 years on a permanent basis.

The specific proposed changes are listed in the following section.

2.0 PROPOSED CHANGES

The containment leakage rate testing program in Technical Specification 5.5.15 currently says

> "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 exception:
>
> > ANS 56.8-1994, Section 3.3.1: WCCPPS isolation valves are not Type C tested."

The proposed TS 5.5.15 is as follows:

> 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 NEI 94-01, Revision 3A, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012, as modified by the following exception:
>
> > ANS 56.8-2002, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

3.0 BACKGROUND

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

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

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

By letter dated January 13, 1997, Indian Point Unit 3 submitted a TS change that was supplemented by letters dated March 24, 1997, May 13, 1997 and May 23, 1997 requesting the implementation of 10 CFR 50, Appendix J, Option B. The NRC approved this request as Amendment 174 issued in NRC letter of June 17, 1997. The NRC noted the proposed TS changes were in compliance with the requirements of Option B, and are consistent with the guidance in RG 1.163. With the approval of the amendment, IP3 transitioned to a performance-based ten year frequency for the Type A tests.

Entergy submitted an Amendment request to extend the ILRT interval one time from ten years to 15 years in a letter dated September 6, 2000 that was supplemented by letters dated January 18, 2001 and April 2, 2001. This one-time extension was approved by the NRC, as license amendment 206 on April 17, 2001.

By letter dated August 31, 2007, NEI submitted NEI 94-01, Revision 2, and EPRI report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," to the NRC Staff for review. NEI 94-01, Revision 2, describes an approach for implementing the optional performance-based requirements of Option B, which includes provisions for extending Type A intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. This method uses industry performance data, plant-specific performance data, and risk insights in determining the appropriate testing frequency. NEI 94-01, Revision 2, also discusses the performance factors that licensees must consider in determining test intervals.

The NEI guideline does not address how to perform the tests because these details are included in referenced industry documents (e.g., American National Standards institute/American Nuclear Society (ANSI/ANS) 56.8-2002).

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

EPRI Report No. 1009325, Revision 2, provides a risk impact assessment for optimized ILRT intervals of up to 15 years, using current industry performance data and risk-informed guidance, primarily Revision 1 of RG 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The NRC's final SE issued by letter dated June 25, 2008, documents the evaluation and acceptance of EPRI Report No. 1009325, Revision 2, subject to the specific limitations and conditions listed in Section 4.2 of the SE. An accepted version of EPRI Report No. 1009325 has subsequently been issued as Revision 2-A (also identified as Technical Report TR-1018243) dated October 2008.

The proposed amendment would revise TS 5.5.14, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," with a reference to Nuclear Energy institute (NEI) topical report NEI 94-01, "Industry Guideline for implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, as the implementation document used by Entergy to develop the Indian Point 3 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J (Option B).

Revision 2-A of NEI 94-01 describes an approach for implementing the optional performance-based requirements of Option B, including provisions for extending primary containment integrated leak rate test (ILRT) intervals to 15 years, and incorporates the regulatory positions stated in RG 1.163. In the SE issued by NRC letter dated June 25, 2008, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE. Revision 3-A of NEI 94-01 describes an approach for implementing the optional performance-based requirements of Option B, including provisions for extending primary containment ILRT intervals to 15 years, and also provides for the performance based extension of Type C test intervals for up to 75 months (rather than the 60 months allowed by Regulatory Guide 1.163) with guidance for implementing the extended intervals. In the SE issued by NRC letter dated June 8, 2012, the NRC concluded that NEI 94-01, Revision 3, describes an acceptable approach for implementing the optional performance-based requirements of Option B, and found that NEI 94-01, Revision 3, is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE. Both of these limitations and conditions apply when extending Types C leakage testing beyond 60 months. IPEC is not applying for the extended Type C performance based testing beyond 60 months.

The proposed extension of the interval for the primary containment ILRT, which is currently required to be performed at ten year intervals, to 15 years from the last ILRT would revise the next scheduled ILRT to March 2020 as opposed to the ILRT currently scheduled for March 2015. This is approximately 15 years since the last ILRT which was completed March 2005. The prior ILRT was performed December 1990 and had obtained a one-time extension to the ten-year ILRT frequency in license amendment no. 206, issued April 17, 2001. The currently proposed change would allow successive ILRTs to be performed at 15-year intervals (assuming acceptable performance history). The performance of fewer ILRTs would result in significant savings in radiation exposure to personnel, cost, and critical path time during future refueling outages.

4.0 Technical Evaluation

As required by 10 CFR 50.54(o), the IP3 containment is subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J which requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the 10 CFR 50 Appendix J Testing Program Plan is based on RG 1.163, which endorses NEI 94-01, Revision 0. This LAR proposes to revise the 10 CFR 50, Appendix J Testing Program Plan by implementing the guidance in NEI 94-01, Revision 3-A but will not extend the Type C leakage beyond 60 months.

4.1 Limitations and Conditions

In the June 25, 2008 NRC SE, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE.

The following Table 4.1 – 1 lists the SE Section 4.1 Limitations and Conditions as well as compliance with each of the six limitations and conditions.

Table 4.1-1

Limitations and Conditions (Section 4.1 of Safety Evaluation Dated June, 25,2008)	**IP3 Compliance**
For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SE Section 3.1.1.1).	Implementation of NEI 94-01 Rev 3A will require use of the definition of "performance leakage rate" defined in Section 5.0 for calculating the Type A leakage rate when performing Type A tests.
The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3).	NEI-94-01 3A, Section 9.2.3.2 requires a general visual examination prior to each Type A test and at least 3 other outages before the ILRT. This should be scheduled in conjunction with or coordinated with examinations required by ASME Code, Section XI, Subsections IWE and IWL. A schedule of containment inspections is provided in Section 4.4
The licensee addresses the areas of the containment structure potentially	A general visual examination of accessible interior and exterior surfaces is conducted per

Table 4.1-1

Limitations and Conditions (Section 4.1 of Safety Evaluation Dated June, 25,2008)	IP3 Compliance
subjected to degradation. (Refer to SE Section 3.1.3).	the Containment Inservice Inspection Plan which implements the requirements of ASME, Section XI, Subsections IWE and IWL. IP3 will explore / consider inaccessible degradation-susceptible areas that can be inspected using viable, commercially available NDE methods.
The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4).	The design change process will address any testing and inspection requirements following future major modifications to the containment structure. This process provides a disciplined approach for determining the program and system interfaces associated with design change. This process evaluates requirements pertaining to the ASME Containment In-Service Inspection Program, ASME Appendix J (Primary Containment Leak Rate Testing) Program, and ASME Section XI
The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2).	IP3 is adopting, consistent with Section 9.2.2 of NEI 94-01, a Type A test interval defined as the time period from the completion of a Type A test to the start of the next test. This definition will be used for scheduling and planning of the next Type A test to the month and year (see RIS 2008-27).
For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable to IP3.

In the June 8, 2012 NRC SE, the NRC concluded that NEI 94-01, Revision 3, describes an acceptable approach for implementing the optional performance-based requirements of Option B (Type B and C tests were addressed), and found that NEI 94-01, Revision 3, is acceptable for referencing by licensees proposing to amend their TS in regard to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.1 of the SE.

1. Condition 1

   The approved NEI 94-01 is allowing Type C LLRTs to be increased to 75 months with the permissible extension to 84 months for non-routine emergent conditions. This is subject to certain exceptions.

2. Condition 2

   The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as found minimum pathway leakage rates for the just tested penetrations are summed with the as-left minimum pathway leakage rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Given the required margin included with the performance criterion and the considerable extra margin most plants consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of trending.

Since the 60 month limitation on Type C penetrations is being retained, these conditions are not applicable and do not have to be addressed

4.2 Existing Exceptions

The provisions of RG 1.163 have been incorporated into NEI 94-01 Revision 3A so the exception to RG 1.63 will remain unchanged except to incorporate the change in ANSI standards (refers to ANS 56.8-2002 rather than ANS 56.8-1994). The revised exception will say "ANS 56.8-2002, Section 3.3.1: WCCPPS isolation valves are not Type C tested."

4.3 Previous Test results

4.3.1 ILRT Test Results

Past IP3 ILRT results have confirmed that the containment is acceptable with respect to the design criterion of 0.1% leakage of containment air weight at the design basis loss of coolant accident pressure (La). Since the last two Type A "as found" tests for IP3 had "as found" test results of less than 1.0La, a test frequency of 15 years in accordance with NEI 94-01 Revision 3A would be acceptable. The last two tests were:

Last ILRT in March 2005

The measured containment leak rate (Ltm) at the test pressure of 60.61 psia was 0.0565 % containment air weight / day with a 95% confidence level.

Prior ILRT in December 1990

The measured containment leak rate (Ltm) at the test pressure of 59.49 psia was 0.032 % containment air weight / day with a 95% confidence level.

For background, the prior three Type A tests had the following results:

Date	As found Leakage (% Containment weight per day)	Test Pressure (psia)
July 27, 1987	0.34(1)	59.89
August 4, 1982	0.034	60.00
August 2, 1978	0.14(2)	60.00

Notes: (1) There was a leak through the reactor coolant pump seal water return valve MOV-222 on penetration R, Line 17. Valve 221A was closed to isolate MOV-222 and the leakage returned to normal.

(2) There was a leak in the #33 and #34 containment fan cooler service water supply and return lines inside containment.

4.3.2 Type B and C testing

The IP3 Appendix J, Type B and Type C testing program requires testing of the components required by 10 CFR 50, Appendix J, Option B. Technical Specification Amendment 174, dated June 17, 1997, approved the adoption of 10 CFR 50, Appendix J, Option B performance based testing requirements for containment leakage testing. The minimum pathway combined Type B and Type C leakage from the March 2005 outage, when the last Type A test was performed, is provided below. The subsequent combined as found Type B and Type C test values during each successive outage since the last Type A test are also provided below. The data is provided in percentage of leakage allowed (0.6La).

Table 4.3-1

Date	**As-Found Leakage (sccm)**	**La (ccm)**	**Percent ((As-Found/La) x100)**	**Percent ((As-Found/.6La))x100)**
April 2005	41585	119689	0.208	0.347
April 2007	30352	119689	0.152	0.254
April 2009	44621	119689	0.224	0.373
April 2011	51878	119689	0.260	0.433
March 2013	36669	119689	0.184	0.306

Based on the results discuss the largest as found leakage and the as left conditions are within the acceptance criterion associated with the 15 year ILRT.

Table 4.3-2 provides a listing of the containment penetrations subject to Type B and C testing, the test frequency, the last test date and the next test date, and the as left leakage. Notes are provided for test failures.

Table 4.3-2						
Penetration	Description	Type	Test Frequency (months)	Last Test date	Next test date	"as – Left" Leakage (cc/min)
	Fuel Transfer Tube	B	30	3/27/13	3/15	48
	Equipment Hatch Seal	B	30	3/27/13	3/15	635
	Personnel Airlock – 80 foot	B	30	9/26/13	9/15	3076
	Personnel Airlock – 95 foot	B	30	9/26/13	9/15	3212
	WCCPP Zone 2 - Racks 10, 11	B(1)	36	1/3/12	1/28/14	30,821.70
	WCCPP Zone 2 – Racks 12, 13	B(1)	36	1/3/12	1/28/14	1028.75
Y	Pressurizer relief tank N_2 supply tank RCS – Valve RC-518	C	30	3/8/13	3/15	86
Y	Pressurizer relief tank N_2 supply tank RCS – Valve RC-550	C	60	3/8/13	3/17	21.75
GG	Containment spray headers – Valves SI-867A, SI-878A	C	60	3/14/13	3/17	64
P	Containment spray headers – Valves SI-867B, SI-878B	C	60	3/2/13	3/15	79.8
RR	Accumulator N2 supply – Valve NNE-	C	30	3/9/13	3/15	37
RR	Accumulator N2 supply – Valve NNE-863	C	60	3/9/13	3/15	0
V	Primary system vent and N2 supply – Valve WD-1610	C	60	3/12/11,	3/15	
V	Primary system vent and N2 supply – Valve WD-1616	C	30	3/12/13	3/15	0
RR	Containment Air Sample In (Rad) – Valves PCV-1234, PCV-1235	C	60	3/19/13	3/17	7
RR	Containment Air Sample In (Rad) – Valves PCV-1236, PCV-1237	C	60	3/19/13	3/17	1
R	Air Ejector Discharge to Containment - Valves CA-1229, CA-1230	C	30	2/26/13	8/26/15	142
EE	Vent Purge Supply Duct - Valves VS-1170, VS-1171	C(1)	30	3/24/13	3/15	1620.0

Table 4.3-2						
Penetration	Description	Type	Test Frequency (months)	Last Test date	Next test date	"as – Left" Leakage (cc/min)
FF	Vent Purge Exhaust Duct - Valves VS-1172, VS-1173	C(1)	30	3/25/13	3/15	6842.5
PP	Cont Pressure Relief Vent - Valves VS-1190, VS-1191, VS-1192	C(1)	30	3/23/13	3/15	432.8
R, TT, LL	Post Accident Sample system supply lines - Valves SP-506, SP-507, SP-509	C	30	3/10/13	3/15	535
Z, O, Z	Post Accident Sample system supply lines - Valves SP-512, SP-513, SP-514	C	30	3/10/13	3/15	945.5
O	Post Accident Sample system return lines - Valves SP-510, SP-511	C	30	3/10/13	3/15	867.5
R	Post Accident Sample system return lines - Valves SP-515, SP-516	C	30	3/10/13	3/15	267
Y	instrument air (post accident vent supply) - Valve IA-39	C	30	3/12/13	3/15	52.25
Y	instrument air (post accident vent supply) - Valve IA-1228	C	60	3/29/11	3/15	21.75
	Equipment access – Valves CB-5, CB-6	C	30	2/28/13	3/15	248
	Equipment access – Valve CB-7	C	60	2/28/13	3/15	287
	Personnel air lock – Valves CB-1, CB-2	C	30	2/28/13	3/15	70
	Personnel air lock – Valve CB-3	C	60	2/28/13	3/17	109

Note 1 – The Weld Channel and Penetration Pressurization System is controlled by Technical Specification 3.6.10 which requires a surveillance every 36 months to demonstrate the leakage rate for the WC&PPS is $\leq$ 0.2% of the containment free volume per day when pressurized to $\geq$ 43 psi above containment pressure. The leakage rate from this test is included in the containment leakage summary for the electrical penetrations and the piping penetrations. When the test was performed the leakage was high and CR-2012-741 was written. It noted there was 84,185cc/min leakage for Weld Channels Zone I and Zone II, respectively. These Zones test the electrical and mechanical penetrations as well as other components already Type C tested (e.g., Containment Purge supply and exhaust, equipment hatch, Containment pressure relief, personnel hatch, Containment Pressure Relief) whose leakage was removed from the total.

4.4 Code Inspections

Prior to each Type A test a general visual examination is required of accessible interior and exterior surfaces of the containment for structural issues that may affect the performance of the Type A test. This inspection will be performed as part of the Containment Inservice Inspection (ISI) Plan to implement the requirements of ASME, Section XI, Subsection IWE and IWL (the applicable code edition and addenda for the fourth 10 year interval is ASME Section XI, 2001 Edition including the 2002 and 2003 Addenda in paragraph (b)(2)).

The examination performed in accordance with the ISI program to meet Subsections IWE and IWL satisfies the general visual examination requirements specified in Option B. The identification and evaluation of inaccessible areas are addressed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix). Each ten year ISI interval is divided into three approximately equal inspection periods. A minimum of one inspection required by the IWE inspection program is performed during each inspection period of the ISI period to meet the program requirements. IWL visual examinations of accessible concrete containment surfaces are to be completed once every 5 years within the limitations specified in IWL-2410(b), (c), and (d) resulting in at least two IWL examinations being performed during a 15 year type A and typically scheduled in two of the three inspection periods of a 10 year ISI interval. Therefore, the frequency of the examinations performed in accordance with the IWE / IWL program will satisfy the requirements of NEI 94-01 Revision 3A, Section 9.2.3.2, to perform a general visual examination before the Type A test during at least three other outages before the next Type A test if the interval is extended to 15 years. The last ILRT was performed March 2005 and the next 15 year interval will end in 3R20 scheduled for the spring of 2019. The following Table illustrates the current and planned inspection intervals for the IP3 first and second IWE / IWL inspection intervals:

Table 4.4-1
IWE / IWL Inspections

Inspection Interval	Inspection Period	Period Start Date	Period End Date	Refuel Outage	Refuel Month/Year
1	1	September 10, 1999	July 20, 2003	3R11 3R12	Spring 2001 Spring 2003
1	2	July 21,2003	July 20, 2006	3R13	Spring 2005
1	3	July 21, 2006	July 20, 2009	3R14 3R15	Spring 2007 Spring 2009
2	1	July 21, 2009	July 20, 2013	3R16 3R17	Spring 2011 Spring 2013
2	2	July 21, 2013	July 20, 2016	3R18	Spring 2015
2	3	July 21, 2016	July 20, 2019	3R19 3R20	Spring 2017 Spring 2019

The following information provides the IP3 IWE examination results of the containment metal liner completed during refuel outages 3R15 (2009) and 3R17 (2013) and the IWL examination results for the containment concrete visual inspections completed in 2005 and 2009 (these are not always completed in an outage). The next IWE examination is scheduled for 2015 and the inspection will also be scheduled for 3R20 (2019) prior to the proposed date for the next ILRT. The next IWL examination is scheduled for 2014 and the inspection will also be scheduled prior to the proposed date for the next ILRT 3R20 (2019). Corrective Actions identified by these inspections are provided with the discussions. There are no primary containment surface areas that require augmented examination in accordance with ASME Section XI, IWE-1240.

4.4.1 IWE Examinations

Refueling Outage 3R15 (2009) Containment Inservice Inspection – Metal Liner

Examinations were performed for the Containment Liner Plate during 3R15 in 2009. There were deficiencies noted on the 46' elevation of the Containment Building; most notably pitting and corrosion in the stainless steel insulation jacket, missing caulking in vapor barrier, missing/loose attachments and rusting/peeling paint. These conditions were documented in IP3 Corrective Action Program under Condition Report CR-IP3-2009-00805.

The caulking in the vapor barrier was repaired. The areas of pitting and corrosion of the stainless steel jacket was caused by past water leaks emanating from the Fan Cooler units located above the location of corrosion. The peeling paint on penetrations was observed by Civil Engineering during walk downs. While the penetrations did exhibit some surface corrosion, no flaking or material loss was observed. The penetrations were considered structurally adequate with no repairs needed at the time.

All of the conditions noted during this inspection did not result in any structural degradation that adversely affects the ability of the containment to perform its design function of maintaining integrity during accident conditions.

Refueling Outage 3R17 (2013) Containment Inservice Inspection – Metal Liner

Examinations performed for the Containment Liner Plate identified several small areas that were recorded as indications on the NDE examination reports and documented in the Corrective Action Program as Condition Report CR-IP3-2013-01382 for tracking. Per the NDE examiner, these indications were mostly peeling paint, surface scratches, and light rust. All flaking paint on the containment dome liner had been identified in report IP3-RPT-STR-2968 and NDE-80-3-006-CB and tracked by GSI-191. There were no indications of leakage detected surrounding the mechanical or electrical penetrations. The condition of the stainless steel insulation covering some containment liner plates has no indications since the 2009 inspection.

All reported visual observations were considered minor with no areas of suspect damage or deterioration which would impact the structural integrity or leak tightness of the containment liner. The NDE report results were noted to be Accepted.

4.4.2 IWL Examinations

The IWL examinations are general visual inspections of Class CC components and the Reinforced Concrete shell of Class CC pressure retaining components of the Vapor Containment (VC) for Unit 3 at Indian Point Energy Center. The inspections are performed to identify signs of structural degradation that may affect the structural integrity or leak tightness and to identify the required repairs and/or replacement activities to minimize degradation due to environmental conditions and aging. The last two inspections were performed in 2005 and 2009.

The inspections are general visual inspections performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, 1998 Edition, Section XI. Division 1, Subsection IWL as required and modified by NRC, Code of Federal Regulation, Title 10, Part 50, Section 55a, "Codes and Standards,"(10 CFR 50.55a — 1999). When needed, optical enhancement equipment with zoom capabilities are used as visual aids during the inspections.

All of the inspections are performed under the direction of the IWL Responsible Engineer (RE). The RE is the Civil/Structural Design Engineering Supervisor at IPEC and a New York State Registered Professional Engineer in accordance with the IWL Procedure. The Responsible Engineer has knowledge of the Design and Construction Codes as well as other criterion used in IP3's Containment. Degreed engineers perform the inspections under the direction of the RE and are knowledgeable and trained in the design, evaluation and performance requirements of structures and qualified to perform visual examination either directly or remotely, with adequate illumination, to detect evidence of degradation.

During the 2009 IWL inspection, several general typical concrete conditions were identified throughout the structure, such as minor cracks and pattern cracking, numerous bugholes, leaching, scaling, and spalling. At this time, none of these indications warrant specific monitoring, thus specific characterization is not required and the conditions will continue to be looked at in the broad sense. Cracks are included as general comments. Reinforced concrete, such as containment, is expected to crack and none of the cracks have any significant width (considered tight) or rust staining. A sample comparison of areas with multiple cracks was made using photos from 2001, 2005 and 2009, and no changes were identified. Therefore, the cracking on containment is concluded to be inactive pending the next integrated leak rate test (ILRT).

The 2001 and 2005 inspections identified several anomalies that were reviewed in the 2009 inspection to determine if any further degradation has occurred. For the most part, the findings from past inspections have remained unchanged over the past cycle. For findings that have further degraded, such as spall areas increasing, they were evaluated and deemed acceptable.

Based on the 2009 inspection, the Containment Structure remains fully capable of performing its design functions. The Concrete Containment is Acceptable with Degradation in accordance with ASME Section XI IWL. The IWL components and structures are capable of performing their structural functions which include protection or support of safety-related systems or components. The components and structures are free of degradation which could lead to possible failure. Indications identified will not deteriorate beyond an acceptable condition prior to the next scheduled inspection. Details of the 2005 and 2009 IWL inspections can be found in reports IP-RPT-06-00013 and IP-RPT-09-011069.

4.5 Confirmatory Analysis

4.5.1 <u>Methodology</u>

An evaluation has been performed to assess the risk impact of extending the IP3 ILRT interval from the current ten years to 15 years. This plant-specific risk assessment followed the guidance in NEI 94-01, Revision 3-A, the methodology outlined in EPRI TR-104285, August 1994 and TR-1009325, Revision 2-A, and the NRC regulatory guidance outlined in RG 1.174 on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change the licensing basis of the plant. In addition, the methodology used for Calvert Cliffs Nuclear Power Plant to estimate the likelihood and risk implication of corrosion-induced leakage of steel containment liners going undetected during the extended ILRT interval was also used for sensitivity analysis.

In their June 25, 2008, SE, the NRC concluded that a 15 year extension to the Type A ILRT interval was acceptable and that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing in a proposal to amend TS to extend the ILRT surveillance interval to 15 years. This approval was subject to the limitations and conditions noted in Section 4.0 of the SE. The following Table 4.5-1 lists the SE Section 4.2 Limitations and Conditions and a description of how the IP3 analysis complies with those four limitations and conditions

Table 4.5 – 1

Limitations and Conditions of Risk Assessment	**IP3 Compliance**
The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.	The technical adequacy of the IP3 PRA and consistency with the RG 1.200 requirements relevant to the ILRT extension are discussed in Section 4.5.2 and detailed in Appendix A of Attachment 3.
The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this SE. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point. While acceptable for this application, the NRC staff is not endorsing these threshold values for other applications. Consistent with this limitation and condition, EPRI Report No. 10 09325 will be revised in the "-A" version of the report, to change the population dose acceptance guidelines and the CCFP guidelines.	The IP3 risk evaluation is summarized in Section 4.5.3 and described in detail in Attachment 3. The results of that evaluation demonstrate that the estimated risk increase is small and consistent with the criteria discussed in the SE.
The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor	The IP3 analysis used a pre-existing containment leak rate of 100La to calculate the increase in population dose for the large leak rate accident case (EPRI Class 3b) . (Attachment 3, Section

Table 4.5 – 1	
Limitations and Conditions of Risk Assessment	**IP3 Compliance**
operation). In order to make the methodology acceptable, the average leak rate for the pre-existing containment large leak rate accident case (accident case 3b) used by the licensees shall be 100 La instead of 35 La.	1.3).
A LAR is required in instances where containment over-pressure is relied upon for ECCS performance.	Containment overpressure is not relied upon for ECCS performance (Attachment 3, Section 5.8).

4.5.2 PRA Quality

The risk assessment performed for the IP3 ILRT extension request is based on the current Level 1 and Level 2 PRA model of record, which was released in November 2012. Information developed for the license renewal effort to support the Level 2 release categories is also used in this analysis supplemented by additional calculations to more appropriately represent the intact containment case in the ILRT extension risk assessment. A discussion of the Entergy model update process, the peer review performed on the IP3 model, the results of that peer review and the potential impact of peer review findings on the ILRT extension risk assessment are provided in Attachment 3, Section A.2.

It should be noted that, while the analysis presented in Attachment 3 was performed for both IP2 and IP3, this submittal only addresses a LAR for IP3. The IP2 information presented in Attachment 3 is therefore informational only and not part of the basis for the current LAR.

4.5.3 Summary of Plant-Specific Risk Assessment Results

The findings of the IP3 risk assessment confirm the general findings of previous studies that the risk impact associated with extending the ILRT interval to one in 15 years is small. The IP3 plant-specific results for extending the ILRT interval to 15 years, taken from Attachment 3, Section 7.0, Conclusions, are summarized below.

1. Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines "very small" changes in risk as resulting in increases of CDF below 1.0E-06/yr and increases in LERF below 1.0E-07/yr. "Small" changes in risk are defined as increases in CDF below 1.0E-05/yr and increases in LERF below 1.0E-06/yr. Since the ILRT extension was demonstrated to have no impact on CDF for IP3, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval for the base case with corrosion included for IP3 is estimated at 1.26E-07/yr (see Attachment 3, Table 5.6-1b), which is within the small change region of the acceptance guidelines in Reg. Guide 1.174. In using the EPRI Expert Elicitation methodology, the change is estimated as 1.34E-08/yr (see Attachment 3, Table 6.2-2b), which is within the very small change region of the acceptance guidelines in Reg. Guide 1.174.

2. The change in dose risk for changing the Type A test frequency from three-per-ten years to

once-per-fifteen-years, measured as an increase to the total integrated dose risk for all internal events accident sequences for is 0.751 person-rem/yr (0.93%) using the EPRI guidance with the base case corrosion case (Attachment 3, Table 5.6-1b). The change in dose risk drops to 0.142 person-rem/yr when using the EPRI Expert Elicitation methodology (Attachment 3, Table 6.2-2b).

3. The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen years including corrosion effects using the EPRI guidance (see Section 5.5) is 0.85% for IP3. This value drops to less that 0.10% for IP3 using the EPRI Expert Elicitation methodology (see Attachment 3 Table 6.2-2b). This is below the acceptance criteria of less than 1.5% defined Attachment 3 in Section 1.3.

4. To determine the potential impact from external events, a bounding assessment from the risk associated with external events utilizing information from the IP3 IPEEEs similar to the approach used in the License Renewal SAMA analysis. As shown in Attachment 3 Table 5.7-2b the total increase in LERF for IP3 due to internal events and the bounding external events assessment is 5.70E-07/yr. This value is in Region II of the Reg. Guide 1.174 acceptance guidelines.

5. As shown in Attachment 3, Table 5.7-4, the same bounding analysis indicates that the total LERF from both internal and external risks is 6.35E-06/yr for IP3, which is less than the Reg. Guide 1.174 limit of 1.0E-05/yr given that the ΔLERF is in Region II (small change in risk).

6. Finally, since the external events assessment led to exceeding one of the two alternative acceptance criteria (*i.e. greater than 1.0 person-rem/yr, an alternative detailed bounding* external events assessment was also performed to demonstrate that the alternate 1.0% person-rem/yr criterion and the other acceptance criteria could still be met. In this case, as shown in Attachment 3, Table 5.7-7 for IP3, the total change in LERF from both internal and external events was 5.97E-7/yr, the change in person-rem/yr was 3.55/yr representing 0.65% of the total, and the change in the CCFP was 0.89%. All of these calculated changes meet the acceptance criteria. As shown in Attachment 3, Table 5.7-8, this assessment indicates that the total LERF from both internal and external risks is 2.83E-06/yr for IP3, which is less than the Reg. Guide 1.174 limit of 1.0E-05/yr given that the ΔLERF is in Region II (small change in risk).

7. Including age-adjusted steel liner corrosion effects in the ILRT assessment was demonstrated to be a small contributor to the impact of extending the ILRT interval for IP3.

Therefore, increasing the ILRT interval on a permanent basis to a one-in-fifteen year frequency is not considered to be risk significant. Details of the IP3 risk assessment are contained in Attachment 3.

4.6 Conclusion

NEI 94-01, Revision 3-A, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. IP3 is proposing to adopt the guidance of NEI 94-01, Revision 3-A for the 10 CFR 50, Appendix J, testing program plan.

Based on the previous ILRT tests conducted at IP3, supplemented by risk analysis studies, including the IP3 risk analysis provided in Attachment 3, it may be concluded that extension of the containment ILRT interval from ten to 15 years represents minimal risk performed in accordance with Option B and inspected per the guidance NEI-94-01 Revision 3A.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy has evaluated the safety significance of the proposed change to the IP3 TS which revise IP3 TS 3.5.15, "Containment Leakage Rate Testing Program," to allow a permanent extension to the frequency of Type A testing based upon performance criteria. The proposed changes have been evaluated according to the criteria of 10 CFR 50.92, "Issuance of Amendment". Entergy has determined that the subject changes do not involve a Significant Hazards Consideration, 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 amendment involves changes to the IP3 containment leakage rate testing program. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary 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 do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

   The proposed amendment adopts the NRC accepted guidelines of NEI 94-01, Revision 3-A, for development of the IP3 performance-based testing program for the Type A testing. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components would limit leakage rates to less than the values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in RG 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. Entergy has determined that the increase in conditional containment failure probability due to the proposed change would be very small. Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated.

Therefore, 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 amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, for the development of the IP3 performance-based leakage testing program, and establishes a 15-year interval for the performance of the containment ILRT. 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 do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

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

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

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, for the development of the IP3 performance-based leakage testing program, and establishes a 15-year interval for the performance of the containment ILRT. This amendment does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the containment leakage rate testing program, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests would be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 3-A.

Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment would not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current IP3 PSA model concluded that extending the ILRT test interval from ten years to 15 years results in a very small change to the risk profile.

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

Based on the above, Entergy concludes that the proposed amendment to the Indian Point 3 Technical Specifications presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of 'no significant hazards consideration' is justified.

5.2 Applicable Regulatory Requirements / Criteria

The NRC Order of February 11, 1980 required an evaluation of the degree of compliance with the GDC at the time. This section discusses continued compliance with certain of those criteria.

The plant will continue to meet Criterion 1 of 10 CFR 50.36 which says "Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems and components important to safety shall be maintained by or under the control of the nuclear power plant licensee throughout the life of the unit" and Criterion 3 which says "Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed."

The extension of the duration of the ILRT for the containment will not affect the design, fabrication, or construction of the containment structure and the design will continue to account for the effects of natural phenomena. The ILRT of the containment will continue to be done in accordance with 10 CFR 50 Appendix J using 10 CFR 50 Appendix B quality standards. The frequency of the ILRT is being changed in accordance with standards reviewed and approved as compliant with Appendix J. Therefore there will be no instances where the applicable regulatory criteria are not met.

5.3 Environmental Considerations

The proposed changes to the IP3 TS do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

This request is similar in nature to the license amendment authorized by the NRC on April 22, 2012 for the Palisades Nuclear Plant (TAC No. ME5997, Accession Number ML120740081).

ATTACHMENT 2 TO NL-14-014

MARKED UP TECHNICAL SPECIFICATIONS PAGES FOR PROPOSED CHANGES REGARDING 15 YEAR CONTAINMENT ILRT

Changes indicated by lineout for deletion and Bold/Italics for additions

Unit 3 Affected Pages:

5.5-30

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

5.5 Programs and Manuals

5.5.15 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 ***NEI 94-01, Revision 3A, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 12, 2012,*** ~~the guidelines contained in Regulatory Guide 1.163, "Performance Based Containment Leak Test Program, dated September 1995"~~ as modified by the following exception:

ANS 56.8-~~1994~~2002, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

The maximum allowable primary containment leakage rate, L_a, at a minimum test pressure equal to P_a, shall be 0.1% of primary containment air weight per day. P_a is the peak calculated containment internal pressure related to the design basis accident.

Leakage acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

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

2) For each door, leakage rate is $\leq$ 0.01 L_a when pressurized to $\geq$ P_a,

c. Isolation Valve Seal Water System leakage rate acceptance criterion is $\leq$ 14,700 cc/hr at $\geq$ 1.1 P_a.

d. Acceptance criterion for leakage into containment from isolation valves sealed with the service water system is $\leq$ 0.36 gpm per fan

(continued)

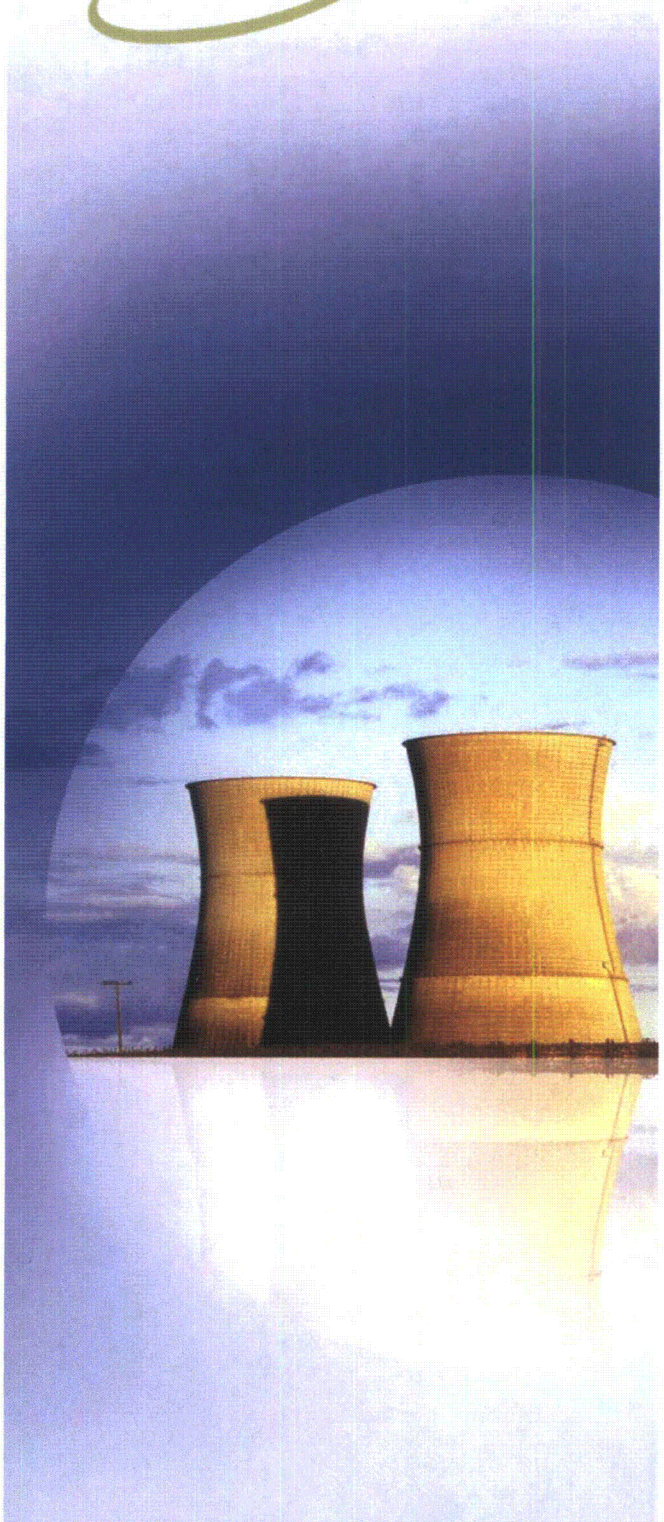
ATTACHMENT 3 TO NL-14-014

RISK IMPACT OF EXTENDING THE ILRT INTERVAL ASSOCIATED WITH THE PROPOSED TECHNICAL SPECIFICATION CHANGES

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

an SKF Group Company

RISK ASSESSMENT FOR INDIAN POINT REGARDING THE ILRT (TYPE A) PERMANENT EXTENSION REQUEST



Prepared for:

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October 2013

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RISK ASSESSMENT FOR INDIAN POINT REGARDING THE ILRT (TYPE A) PERMANENT EXTENSION REQUEST

Revision 0

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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 permanent extension of the Indian Point Units 2 and 3 (IP2 and IP3) containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology outlined in EPRI TR-104285 [2], the EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [3], 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 [5]. 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 [3].

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 the normal containment leakage of 1.0La (allowable leakage).

The basis for a 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 [6], "Performance-Based Containment Leak Test Program," 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 [2].

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Because ILRTs represent substantial resource expenditures, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures to support a reduction in the test frequency for IP2 and IP3.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 [2] methodology to perform the risk assessment. In October 2008, EPRI 1018243 [3] was issued to develop a generic methodology for the risk impact assessment for ILRT interval extensions to 15 years using current performance data and risk informed guidance, primarily NRC Regulatory Guide 1.174 [4]. This more recent EPRI document considers the change in population dose, large early release frequency (LERF), and containment conditional failure probability (CCFP), whereas EPRI TR-104285 considered only the change in risk based on the change in population dose. This ILRT interval extension risk assessment for IP2 and IP3 employs the EPRI 1018243 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 permanent 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 1.0E-06 per reactor year and increases in large early release frequency (LERF) less than 1.0E-07 per reactor year. Note that a separate discussion in Section 5.8 confirms that the CDF is not impacted by the proposed change for IP2 and IP3. Therefore, since the Type A test does not impact CDF for IP2 and IP3, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 1.0E-06 per reactor year, provided that the total LERF from all contributors (including external events) can be reasonably shown to be less than 1.0E-05 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) is also calculated to help ensure that the defense-in-depth philosophy is maintained.

With regard to population dose, examinations of NUREG-1493 and Safety Evaluation Reports (SERs) for one-time interval extension (summarized in Appendix G of [3]) indicate a range of incremental increases in population dose[1] that have been accepted by the NRC. The range of

[1] The one-time extensions assumed a large leak (EPRI class 3b) magnitude of 35La, whereas this

incremental population dose increases is from ≤0.01 to 0.2 person-rem/yr and 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (Figure 7-2 of NUREG-1493) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, the NRC SER on this issue [7] defines a small increase in population dose as an increase of ≤ 1.0 person-rem per year, or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This definition has been adopted by the IP2/IP3 analysis.

The acceptance criteria are summarized below.

1. The estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years must be demonstrated to be small. (Note that Regulatory Guide 1.174 defines very small changes in risk as increases in CDF less than 1.0E-6 per reactor year and increases in LERF less than 1.0E-7 per reactor year. Since the type A ILRT test is not expected to impact CDF for Indian Point, the relevant risk metric is the change in LERF. Regulatory Guide 1.174 also defines small risk increase as a change in LERF of less than 1.0E-6 reactor year.) Therefore, a small change in risk for this application is defined as a LERF increase of less than 1.0E-6.
2. Per the NRC SE, a small increase in population dose is also defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive.
3. In addition, the SE notes that a small increase in Conditional Containment Failure Probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests (typically about 1% or less, with the largest increase being 1.2%). This would require that the increase in CCFP be less than or equal to 1.5 percentage points.

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analysis uses 100La.

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI methodology is used for evaluating the change in risk associated with increasing the test interval to fifteen years [3]. The analysis uses results from a Level 2 analysis of core damage scenarios from the current IP2 and IP3 PRA analyses of record and the subsequent containment responses to establish the various fission product release categories including the release size.

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 [3].
2. Develop plant-specific population dose rates (person-rem 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 and compare this change with the acceptance guidelines of RG 1.174 [4].
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 variations in the fractional contributions of large isolation failures (due to liner breach) to LERF.

Furthermore,

- Consistent with the previous industry containment leak risk assessments, the IP2 and IP3 assessment uses population dose as one of the risk measures. The other risk measures used in the IP2 and IP3 assessment are the conditional containment failure probability (CCFP) for defense-in-depth considerations, and change in LERF to demonstrate that the acceptance guidelines from RG 1.174 are met.
- This evaluation for IP2 and IP3 uses ground rules and methods to calculate changes in the above risk metrics that are consistent with those outlined in the current EPRI methodology [3].

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The IP2 and IP3 Level 1 and Level 2 internal events PRA models provide representative core damage frequency and release category frequency distributions to be utilized in this analysis.
- It is appropriate to use the IP2 and IP3 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 external events were to be included in the calculations; however, external events have been accounted for in the analysis based on the available information from the IP2 and IP3 IPEEEs [8, 9] as reported and used in the IP2 and IP3 SAMA analysis performed as part of the License Renewal efforts as described in Section 5.7.
- Dose results for the containment failures modeled in the PRA can be characterized by information that was prepared to support the SAMA analysis as part of the License Renewal effort [10]. This information is supplemented with revised calculations [11] for the base case containment intact scenarios which are critical for use in the ILRT extension assessment.
- Accident classes describing radionuclide release end states and their definitions are consistent with the EPRI methodology [3] 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 is 10 La and for Class 3b sequences is 100La, based on the recommendations in the latest EPRI report [3] and as recommended in the NRC SE on this topic [7]. It should be noted that this is more conservative than the earlier previous industry ILRT extension requests, which utilized 35La for the Class 3b sequences.
- Based on the EPRI methodology and the NRC SE, the Class 3b sequences are categorized as LERF and the increase in Class 3b sequences is used as a surrogate for the **Δ**LERF metric.
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The use of the estimated 2035 population data from the MACCS2 off-site consequence runs [10, 11] is appropriate for this analysis. This assumption is consistent with that made in the SAMA analysis.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [12].

4.0 INPUTS

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

4.1 GENERAL RESOURCES AVAILABLE

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

1. NUREG/CR-3539 [13]
2. NUREG/CR-4220 [14]
3. NUREG-1273 [15]
4. NUREG/CR-4330 [16]
5. EPRI TR-105189 [12]
6. NUREG-1493 [6]
7. EPRI TR-104285 [2]
8. Calvert Cliffs liner corrosion analysis [5]
9. EPRI 1018243 [3]
10. NRC Final Safety Evaluation [7]

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 addresses the impact of age-related degradation of the containment liners on ILRT evaluations. EPRI 1018243 complements the previous EPRI report and provides the results of an expert elicitation process to determine the relationship between pre-existing containment leakage probability and magnitude. Finally, the NRC Safety Evaluation (SE) documents the acceptance by the NRC of the proposed methodology with a few exceptions. These exceptions (associated with the ILRT Type A tests) were addressed in the Revision 2-A of NEI 94-01 and the final version of the updated EPRI report [3], which was used for this application.

NUREG/CR-3539 [13]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [31] 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 [14]

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

NUREG-1273 [15]

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

NUREG/CR-4330 [16]

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

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

EPRI TR-105189 [12]

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

The result of the study concluded that a small but measurable safety benefit (shutdown CDF reduced by 1.0E-8/yr to 1.0E-7/yr) is realized from extending the test intervals from 3 per 10 years to 1 per 10 years.

NUREG-1493 [6]

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

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [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 [17] Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 [6] 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..."

Release Category Definitions

Table 4.1-1 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology [3]. 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

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.

TABLE 4.1-1
EPRI/NEI CONTAINMENT FAILURE CLASSIFICATIONS

CLASS	DESCRIPTION
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.

Calvert Cliffs Liner Corrosion Analysis [5]

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms was factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. IP2 and IP3 have a similar type of containment.

EPRI 1018243 [3]

This report presents a risk impact assessment for extending integrated leak rate test (ILRT) surveillance intervals to 15 years. 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 and regulatory 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 IP2 and IP3 analysis presented here in Section 6.2.

NRC Safety Evaluation Report [7]

This SE documents the NRC staff's evaluation and acceptance of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, subject to the limitations and conditions identified in the SE and summarized in Section 4.0 of the SE. These limitations (associated with the ILRT Type A tests) were addressed in the Revision 2-A of NEI 94-01 which are also included in Revision 3-A of NEI 94-01 [1] and the final version of the updated EPRI report [3]. Additionally, the SE clearly defined the acceptance criteria to be used in future Type A ILRT extension risk assessments as delineated previously in the end of Section 1.3.

4.2 PLANT-SPECIFIC INPUTS

The IP2 and IP3 specific information used to perform this ILRT interval extension risk assessment includes the following:

- Level 1 and Level 2 PRA model quantification results [18, 19]
- Population dose within a 50-mile radius for various release categories [10, 11]

IP2 and IP3 Internal Events Core Damage Frequencies

The current IP2 and IP3 Internal Events PRA analyses of record are based on an event tree / linked fault tree model characteristic of the as-built, as-operated plant. Based on the results found in Tables J1.6-2 of Reference [18] and Reference [19], the internal events Level 1 PRA core damage frequency (CDF) is 1.17E-05/yr for IP2 and 1.48E-05/yr for IP3.

IP2 and IP3 Internal Events Release Category Frequencies

The Level 2 release category frequencies were developed from the contributions to CDF for those analyzed containment failure modes that were documented in Tables J1.6-2 and Tables J1.7-4 for IP2 and IP3 of Reference [18] and Reference [19], respectively. Table 4.2-1 summarizes the pertinent IP2 and IP3 results in terms of end-states where a representative release category is assigned for each end-state. The total Large Early Release Frequency (LERF) in Table 4.2-1 is 1.16E-06/yr for IP2 and 1.25E-06/yr for IP3. The individual release category frequencies are utilized here to provide the necessary delineation for the ILRT risk assessment with the corresponding EPRI class for each release category. A discussion of the available population dose information for various release categories follows this table.

TABLE 4.2-1
LEVEL 2 RELEASE CATEGORY FREQUENCIES FOR IP2 AND IP3

RELEASE CATEGORY DESCRIPTION	INDIAN POINT 2 (FREQUENCY/YR)	INDIAN POINT 3 (FREQUENCY/YR)
No Containment Failure	7.86E-06	1.13E-05
Late Release	2.71E-06	2.17E-06
Low to Moderate Early Release	4.66E-09	1.17E-07
High Early Release (LERF)	1.16E-06	1.25E-06
LERF: Containment Bypass (SGTR Initiating Events)	9.58E-07	9.19E-07
LERF: Containment Bypass (ISLOCA)	2.77E-08	1.93E-07
LERF: Containment Bypass (Induced SGTR events)	8.72E-08	5.78E-08
LERF: Containment Isolation Failure	1.11E-08	3.99E-09
LERF: Energetic Containment Failures	6.90E-08	7.14E-08
Total:	**1.17E-05**	**1.48E-05**

IP2 and IP3 Population Dose Information

In the License Renewal analysis for IP2 and IP3 [20], the release categories considered the magnitude of the radionuclide release, e.g., concentration of cesium iodide (CsI), and the time of the release. Table 4.2-2 shows how the different release categories were organized for the license renewal effort. While that breakdown was appropriate for that submittal, the breakdown in Table 4.2-1 is sufficient for this ILRT extension risk assessment.

TABLE 4.2-2

RELEASE CATEGORY DEFINITIONS FROM THE LICENSE RENEWAL EFFORT

RELEASE TIMING		RELEASE SEVERITY SOURCE TERM RELEASE FRACTION	
CLASSIFICATION CATEGORY	**TIME OF RELEASE (NOBLE GASES OR CSI)**	**CLASSIFICATION CATEGORY**	**PERCENT CSI IN RELEASE**
Late (L)	> 12 hours	High (H)	> 10
		Moderate (M)	1 to 10
Early (E)	< 12 hours	Low (L)	0.1 to 1
		Low-Low (LL)	0.01 to 0.1
		No Containment Failure (NCF)	< 0.01 (Little to No Release)

The population dose results from latest relevant License Renewal submittal [10] form the basis of the initial ILRT assessment using the latest available release category frequency information as described above. The results for IP2 are taken from Table 5 of Reference [10] and the results for IP3 are taken from Table 6 of Reference [10]. Those population dose results are reproduced in Table 4.2-3 converted to the corresponding values in person-rem (i.e., 100 * person-sv) used for this analysis.

TABLE 4.2-3
POPULATION DOSE PER LICENSE RENEWAL RELEASE CATEGORY FOR IP2 AND IP3

RELEASE CATEGORY DESCRIPTION	INDIAN POINT 2 (PERSON-REM)	INDIAN POINT 3 (PERSON-REM)
No Containment Failure (NCF)	4.75E+03	8.04E+03
Early High	6.51E+07	5.08E+07
Early Medium	1.94E+07	2.00E+07
Early Low	7.93E+06	5.21E+06
Late High	1.63E+07	1.63E+07
Late Medium	6.87E+06	6.85E+06
Late Low	1.61E+06	1.61E+06
Late Low-Low	1.38E+06	1.38E+06

Since the ILRT methodology is based on multipliers to a bounding case which is representative of an allowable leakage of 1.0La, the NCF case from the License Renewal effort, which represents a best estimate release, could not be used. As a result, additional analyses were required for the ILRT assessment to be consistent with the methodology employed. Table 4.2-4 shows the results of four different potential case runs to provide a representative 1.0La release [11]. Note that for the containment intact case, given the similarities between IP2 and IP3, the results are assumed to be applicable to both units. These case results are representative of the 1.0La release as required by the ILRT methodology.

TABLE 4.2-4
POPULATION DOSE FOR INTACT CONTAINMENT CASES FOR IP2 AND IP3

RELEASE CATEGORY DESCRIPTION	INDIAN POINT 2 (PERSON-REM)	INDIAN POINT 3 (PERSON-REM)
Intact Scenario #1 (Vessel Breach Occurs, Containment Fan Coolers Available)	8.28E+04	8.28E+04
Intact Scenario #2 (Vessel Breach Occurs, Containment Sprays Available)	1.59E+04	1.59E+04
Intact Scenario #3 (Vessel Breach Occurs, Fan Coolers and Sprays Available)	1.32E+04	1.32E+04
Intact Scenario #4 (No Vessel Breach, Containment Fan Coolers Available)	2.94E+04	2.94E+04

Based on a review of cutsets associated with the intact containment end state, an apportionment of the intact containment associated release categories was made. First, it was noted that containment sprays were not failed in more than 99% of the intact containment cases for both IP2 and IP3, but their use could only be definitively declared in Medium and Large LOCA scenarios or when vessel breach occurs (i.e., other cases with fan coolers available and no vessel breach are unlikely to reach the automatic containment spray initiation set point of 24 psig for IP2 and 22 psig for IP3). For IP2 about 68% of the intact containment cases also involved no vessel breach, and for IP3 about 63% of the intact containment cases involved no vessel breach. For IP2 and IP3, the medium and large LOCA contribution to the intact containment case was about 10%. Therefore, it was conservatively assumed that just 10% of the intact containment cases could be represented by a case with containment sprays available (i.e., intact scenario #2 from Table 4.2-4). Of the remaining 90%, based on the contribution from no vessel breach scenarios noted above, it was assumed that about 60% of the cases involved scenarios with no vessel failure and about 30% involved scenarios where vessel failure occurred for both IP2 and IP3. Intact scenario #4 from Table 4.2-4 is then used as a representative case for the no vessel failure scenarios, and intact scenario #1 is then conservatively used as a representative case for the remaining vessel failure scenarios. Although sprays are likely available in those scenarios, the SAMG procedures may limit their use based on hydrogen detonation concerns. This leads to an overall weighted average population dose for the intact containment case as shown in Table 4.2-5. This weighted average population dose of 4.41E+04 person-rem is used in the remainder of the calculations using the ILRT methodology.

TABLE 4.2-5
WEIGHTED AVERAGE POPULATION DOSE FOR INTACT CONTAINMENT CASE FOR IP2 AND IP3

RELEASE CATEGORY DESCRIPTION	PERCENT CONTRIBUTION	POPULATION DOSE (PERSON-REM)
Intact Scenario #1 (Vessel Breach Occurs, Containment Fan Coolers Available)	30%	8.28E+04
Intact Scenario #2 (Vessel Breach Occurs, Containment Sprays Available)	10%	1.59E+04
Intact Scenario #3 (Vessel Breach Occurs, Fan Coolers and Sprays Available)	N/A	1.32E+04
Intact Scenario #4 (No Vessel Breach, Containment Fan Coolers Available)	60%	2.94E+04
Weighted Average	0.3 * (8.28E+04) + 0.1 * (1.59E+04) + 0.6 * (2.94E+04)	4.41E+04

<u>Population Dose Risk Calculations</u>

The next step is to take the frequency information from Table 4.2-1, assign each category to the relevant EPRI release category class from Table 4.1-1, and then associate a representative population dose from Table 4.2-3 or Table 4.2-5 for each release category. Table 4.2-6a lists the population dose risk and average population dose organized by EPRI release category for IP2, including the delineation of early and late frequencies for Class 7, and a delineation of SGTR and ISLOCA frequencies for Class 8. Note that the population dose risk (Column 4 of Table 4.2-6a) was found by multiplying the release category frequency (Column 2 of Table

4.2-6a) by the associated population dose (Column 3 of Table 4.2-6a). The corresponding information for IP3 is shown in Table 4.2-6b. Note that only the applicable EPRI release categories at this point are shown in the tables (i.e., the Class 3 frequencies are derived later and the Class 4, 5, and 6 frequencies are not utilized in the EPRI methodology for the ILRT extension risk assessment).

TABLE 4.2-6A
IP2 POPULATION DOSE AND POPULATION DOSE RISK ORGANIZED BY EPRI RELEASE CATEGORY

EPRI RELEASE CATEGORY AND DESCRIPTION	RELEASE FREQUENCY (1/YR)	ASSIGNED POPULATION DOSE (PERSON-REM)	POPULATION DOSE RISK (PERSON-REM/YR)
1: Containment intact	7.86E-06	4.41E+04 [Weighted Average From Table 4.2-5]	3.47E-01
2: Large containment isolation failures	1.11E-08	6.51E+07 [Early High From Table 4.2-3]	7.23E-01
7-CFE: Phenomena-induced containment failures (Early-non LERF)	4.66E-09	1.94E+07 [Early Medium From Table 4.2-3]	9.04E-02
7-CFE: Phenomena-induced containment failures (Early LERF)	6.90E-08	6.51E+07 [Early High From Table 4.2-3]	4.49E+00
7-CFL: Phenomena-induced containment failures (Late)	2.71E-06	6.87E+06 [Late Medium From Table 4.2-3][(1)]	1.86E+01
8-SGTR: Containment bypass (SGTR)	1.05E-06	6.51E+07 [Early High From Table 4.2-3]	6.80E+01
8-ISLOCA: Containment bypass (ISLOCA)	2.77E-08	6.51E+07 [Early High From Table 4.2-3]	1.80E+00
Total:	**1.17E-05**		**94.12**

(1) Although the current model does not distinguish between the different late release categories, the weighted average late release from the License Renewal was within 10% of the Late Medium population dose. The use of the Late Medium population dose for this release category was therefore deemed appropriate for the ILRT assessment.

TABLE 4.2-6B
IP3 POPULATION DOSE AND POPULATION DOSE RISK ORGANIZED BY EPRI RELEASE CATEGORY

EPRI RELEASE CATEGORY AND DESCRIPTION	RELEASE FREQUENCY (1/YR)	ASSIGNED POPULATION DOSE (PERSON-REM)	POPULATION DOSE RISK (PERSON-REM/YR)
1: Containment intact	1.13E-05	4.41E+04 [Weighted Average From Table 4.2-5]	4.98E-01
2: Large containment isolation failures	3.99E-09	5.08E+07 [Early High From Table 4.2-3]	2.03E-01
7-CFE: Phenomena-induced containment failures (Early-non LERF)	1.17E-07	2.00E+07 [Early Medium From Table 4.2-3]	2.34E+00
7-CFE: Phenomena-induced containment failures (Early LERF)	7.14E-08	5.08E+07 [Early High From Table 4.2-3]	3.63E+00
7-CFL: Phenomena-induced containment failures (Late)	2.17E-06	6.85E+06 [Late Medium From Table 4.2-3][(1)]	1.49E+01
8-SGTR: Containment bypass (SGTR)	9.77E-07	5.08E+07 [Early High From Table 4.2-3]	4.96E+01
8-ISLOCA: Containment bypass (ISLOCA)	1.93E-07	5.08E+07 [Early High From Table 4.2-3]	9.80E+00
Total:	**1.48E-05**		**80.96**

(1) Although the current model does not distinguish between the different late release categories, the weighted average late release from the License Renewal was within 10% of the Late Medium population dose. The use of the Late Medium population dose for this release category was therefore deemed appropriate for the ILRT assessment.

The frequencies for the severe accident classes defined in Table 4.1-1 are developed for IP2 and IP3 based on the assignments shown above in Tables 4.2-6a and 4.2-6b. Then, the frequencies for Classes 3a and 3b can be determined with that portion removed from Class 1. This step in the process is described in Section 4.3. Furthermore, adjustments are made to the Class 3b as well as Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4.

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 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 latest EPRI guidance [3], as the mean failure estimated from the available data (i.e., 2 "small" failures that could only have been discovered by the ILRT in 217 tests leads to a 2/217=0.0092 mean value). For Class 3b, consistent with latest available EPRI data, a non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5/(217+1) = 0.0023).

The EPRI methodology contains 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 information includes a discussion of conservatisms in the quantitative guidance for delta LERF. EPRI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The methodology 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 IP2 and IP3 (as detailed in Section 5) means that the Class 2, Class 7, and Class 8 LERF 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. Note that Class 2 events refer to sequences with a large pre-existing containment isolation failure that lead to LERF, a subset of Class 7 events are LERF sequences due to an early containment failure from energetic phenomena, and Class 8 event are containment bypass events that contribute to LERF.

Consistent with the EPRI methodology [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.

<u>IP2 and IP3 Past ILRT Results</u>

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 1.0La, and in compliance with the performance factors in NEI 94-01, Section 11.3. Based on the successful completion of two consecutive ILRTs at IP2 and IP3, 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 noted in the EPRI guidance document [3].

<u>EPRI Methodology</u>

This analysis uses the approach outlined in the EPRI Methodology [3]. The six steps of the methodology are:

1. Quantify the baseline (three-year ILRT frequency) risk in terms of frequency per reactor year for the EPRI accident classes of interest.
2. Develop the baseline population dose (person-rem, from the plant PRA or IPE, or calculated based on leakage) for the applicable accident classes.
3. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.
4. Determine the risk impact in terms of the change in LERF and the change in CCFP.
5. Consider both internal and external events.
6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis.

The first three steps of the methodology deal with calculating 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 fourth step in the 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 IP2 and IP3, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The fourth step of the methodology calculates the change in containment failure probability, referred to as the conditional containment failure probability, CCFP. The NRC has identified a CCFP of less than 1.5% as the acceptance criteria for extending the Type A ILRT test intervals as the basis for showing that the proposed change is consistent with the defense in depth philosophy [7]. As such, this step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174. Step 5 takes into consideration the additional risk due to external events, and Step 6 investigates the impact on results due to varying the assumptions associated with the liner corrosion rate and failure to visually identify pre-existing flaws.

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 [5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. IP2 and IP3 have similar containment types.

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

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

Assumptions

- A half failure is assumed for the basemat concealed liner corrosion due to lack of identified failures.
- The two corrosion events over a 5.5 year data period are used to estimate the liner flaw probability in the Calvert Cliffs analysis and are assumed to be applicable to the IP2 and IP3 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. It is noted that two additional events have occurred in recent years (based on a data search covering approximately 9 years documented in Reference [21]). In November 2006, the Turkey Point 4 containment building liner developed a hole when a sump pump support plate was moved. In May 2009, a hole approximately 3/8" by 1" in size was identified in the Beaver Valley 1 containment liner. For risk evaluation purposes, these two more recent events occurring over a 9 year period are judged to be adequately represented by the two events in the 5.5 year period of the Calvert Cliffs analysis incorporated in the EPRI guidance (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 two years and every ten years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome region, and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability of containment failure versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. For IP2 and IP3, the containment failure probabilities are less than these values at 47 psig, which is the containment design pressure [18, 19]. The probabilities of 1% for the cylinder and dome, and 0.1% for the basemat, albeit conservative, are used in this analysis. Sensitivity studies are included that increase and decrease the probabilities by an order of magnitude (See Table 4.4-1, Step 4).
- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used for the containment cylinder and dome. For the containment basemat, 100% is assumed unavailable for visual inspection. To date, all liner corrosion events have been detected through visual inspection (See Table 4.4-1, Step 5). Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.
- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

TABLE 4.4-1
STEEL LINER CORROSION BASE CASE

<table>
<tr><th>STEP</th><th>DESCRIPTION</th><th colspan="2">CONTAINMENT CYLINDER AND DOME</th><th colspan="2">CONTAINMENT BASEMAT</th></tr>
<tr><td>1</td><td>Historical Steel Liner Flaw Likelihood
Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).</td><td colspan="2">Events: 2

2/(70 * 5.5) = 5.2E-3</td><td colspan="2">Events: 0 (assume half a failure)
0.5/(70 * 5.5) = 1.3E-3</td></tr>
<tr><td rowspan="2">2</td><td rowspan="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).</td><td><u>Year</u>
1
avg 5-10
15</td><td><u>Failure Rate</u>
2.1E-3
5.2E-3
1.4E-2</td><td><u>Year</u>
1
avg 5-10
15</td><td><u>Failure Rate</u>
5.0E-4
1.3E-3
3.5E-3</td></tr>
<tr><td colspan="2">15 year average = 6.27E-3</td><td colspan="2">15 year average = 1.57E-3</td></tr>
<tr><td>3</td><td>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 [5]).</td><td colspan="2">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.)</td><td colspan="2">0.18% (1 to 3 years)
1.04% (1 to 10 years)
2.42% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, values are calculated based on the 3, 10, and 15 year intervals.)</td></tr>
</table>

TABLE 4.4-1
STEEL LINER CORROSION BASE CASE

STEP	DESCRIPTION	CONTAINMENT CYLINDER AND DOME	CONTAINMENT BASEMAT
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the containment cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1% (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4 * 5)	**0.00071% (at 3 years)** =0.71% * 1% * 10% **0.00406% (at 10 years)** =4.06% * 1% * 10% **0.0094% (at 15 years)** =9.40% * 1% * 10%	**0.00018% (at 3 years)** =0.18% * 0.1% * 100% **0.00104% (at 10 years)** =1.04% * 0.1% * 100% **0.00242% (at 15 years)** =2.42% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage that is subsequently added to the EPRI Class 3b contribution is the sum of Step 6 for the containment cylinder and dome, and the containment basemat:

At 3 years : 0.00071% + 0.00018% = 0.00089%

At 10 years: 0.00406% + 0.00104% = 0.00510%

At 15 years: 0.0094% + 0.00242% = 0.01182%

5.0 RESULTS

The application of the approach based on EPRI Guidance [3] has 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.

TABLE 5.0-1
ACCIDENT CLASSES

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION
1	Containment Intact
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 analysis performed examined IP2 and IP3 specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the categorization of the severe accidents contributing to risk was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI 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, if applicable. (EPRI 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 Class 6 sequences). Consistent with the EPRI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.

- Accident sequences involving containment bypass (EPRI Class 8 sequences), large containment isolation failures (EPRI Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI 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.

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 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 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 IP2 and IP3 Level 2 release category frequency results [18, 19]. As described in Section 4.2, the release categories were assigned to the EPRI classes as shown in Table 4.2-6a for IP2 and in Table 4.2-6b for IP3. This application combined with the IP2 and IP3 dose risk (person-rem/yr) also shown in Tables 4.2-6a and 4.2-6b, respectively forms the basis for estimating the increase in population dose risk.

For the assessment of the impact on the risk profile due to the ILRT extension, the potential for pre-existing leaks is included in the model. These pre-existing leak events are represented by the Class 3 sequences in EPRI 1018243 [3]. Two failure modes were considered for the Class 3 sequences, namely Class 3a (small breach) and Class 3b (large breach).

The determination of the frequencies associated with each of the EPRI categories listed in Table 5.0-1 is presented next.

Class 1 Sequences

This group represents the frequency when the containment remains intact (modeled as Technical Specification Leakage). The frequency per year for these sequences is 7.74E-06/yr for IP2 and 1.11E-05/yr for IP3 (refer to Table 5.1-1 for Containment Release Type 1) 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. Note that the values for this Class reported in Table 5.1-1 are slightly lower than that reported in Tables 4.2-6a and 4.2-6b since the 3a and 3b frequencies are now subtracted from Class 1.

Class 2 Sequences

This group consists of large containment isolation failures. For IP2, this frequency is 1.11E-08/yr (refer to Table 5.1-1, Containment Release Type 2). For IP3, this frequency is 3.99E-09/yr (refer to Table 5.1-1, Containment Release Type 2).

Class 3 Sequences

This group represents pre-existing leakage in the containment structure (e.g., containment liner). The containment leakage for these sequences can be either small (2La to 100La) or large (>100La). In this analysis, a value of 10La was used for small pre-existing flaws and 100La for relatively large flaws.

The respective frequencies per year are determined as follows:

$PROB_{Class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.0092 (see Section 4.3)

$PROB_{Class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0023 (see Section 4.3)

As described in Section 4.3, additional consideration is made to not apply these failure probabilities to those cases that are already considered LERF scenarios (i.e., the Class 2, Class 7, and Class 8 LERF contributions). This adjustment is made for based on the frequency information from Tables 4.2-6a and 4.2-6b for IP2 and IP3, respectively as shown below.

For IP2:

Class_3a = 0.0092 * [CDF – (Class 2 + Class 7 LERF + Class 8 SGTR + Class 8 ISLOCA)]
= 0.0092 * [1.17E-05 – (1.11E-08 + 6.90E-08 + 1.05E-06 + 2.77E-08)]
= 9.73E-08/yr

Class_3b = 0.0023 * [CDF – (Class 2 + Class 7 LERF + Class 8 SGTR + Class 8 ISLOCA)]
= 0.0023 * [1.17E-05 – (1.11E-08 + 6.90E-08 + 1.05E-06 + 2.77E-08)]
= 2.43E-08/yr

For IP3:

Class_3a = 0.0092 * [CDF – (Class 2 + Class 7 LERF + Class 8 SGTR + Class 8 ISLOCA)]
= 0.0092 * [1.48E-05 – (3.99E-09 + 7.14E-08 + 9.77E-07 + 1.93E-07)]
= 1.25E-07/yr

Class_3b = 0.0023 * [CDF – (Class 2 + Class 7 LERF + Class 8 SGTR + Class 8 ISLOCA)]
= 0.0023 * [1.48E-05 – (3.99E-09 + 7.14E-08 + 9.77E-07 + 1.93E-07)]
= 3.13E-08/yr

For this analysis, the associated containment leakage for Class 3a is 10La and 100La for Class 3b, which is consistent with the latest EPRI methodology [3] and the NRC SE [7].

<u>Class 4 Sequences</u>

This group represents containment isolation failure-to-seal of Type B test components. 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 this analysis.

<u>Class 5 Sequences</u>

This group represents containment isolation failure-to-seal of Type C test components. Because these failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

<u>Class 6 Sequences</u>

This group is similar to Class 2. These are sequences that involve core damage with a failure-to-seal containment leakage due to failure to isolate the containment. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the EPRI guidance, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences

This group represents containment failure induced by early and late severe accident phenomena. From Table 4.2-6a for IP2, the frequency for early Class 7 sequences is 4.66E-09/yr + 6.90E-08/yr = 7.37E-08/yr, and the frequency for the late Class 7 sequences is 2.71E-06/yr. From Table 4.2-6b for IP3, the frequency for early Class 7 sequences is 1.17E-07/yr + 7.14E-08/yr = 1.88E-07/yr, and the frequency for the late Class 7 sequences is 2.17E-06/yr.

Class 8 Sequences

This group represents sequences where containment bypass occurs (SGTR or ISLOCA). From the frequency information provided in Table 4.2-6a for IP2, the total SGTR contribution to core damage is 1.05E-06/yr and the ISLOCA contribution to core damage is 2.77E-08/yr. From the frequency information provided in Table 4.2-6b for IP3, the total SGTR contribution to core damage is 9.77E-07/yr and the ISLOCA contribution to core damage is 1.93E-07/yr.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to release of radionuclides to the public have been derived in a manner consistent with the definition of accident classes defined in EPRI 1018243 [3] and are shown in Table 5.1-1 for IP2 and for IP3.

TABLE 5.1-1
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (IP2 AND IP3 BASE CASE)

ACCIDENT CLASS (CONTAINMENT RELEASE TYPE)	DESCRIPTION	IP2 FREQUENCY (1/YR)	IP3 FREQUENCY (1/YR)
1	Containment Intact	7.74E-06	1.11E-05
2	Large Isolation Failures (Failure to Close)	1.11E-08	3.99E-09
3a	Small Isolation Failures (liner breach)	9.73E-08	1.25E-07
3b	Large Isolation Failures (liner breach)	2.43E-08	3.13E-08
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A
7-CFE	Failures Induced by Phenomena (Early)	7.37E-08	1.88E-07
7-CFL	Failures Induced by Phenomena (Late)	2.71E-06	2.17E-06
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	1.05E-06	9.77E-07
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	2.77E-08	1.93E-07
CDF	**All CET End States (Including Intact Case)**	**1.17E-05**	**1.48E-05**

5.2 STEP 2 – DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analyses were performed to estimate the weighted average person-rem doses to the population within a 50-mile radius from the plant. The releases are based on a combination of the information provided by the IP2 and IP3 SAMA re-analysis [10], additional population dose runs for the intact containment scenarios [11], and the Level 2 containment failure release frequencies [18, 19] (see Tables 4.2-6a and 4.2-6b of this analysis). The results of applying these releases to the EPRI containment failure classifications are summarized below. Note that the 7-CFE release category is further refined to be the weighted average of the two contributors for moving forward in the ILRT methodology since it is not impacted by the change to the ILRT interval.

For IP2:

Class 1	=	4.41E+04 person-rem (at 1.0La)
Class 2	=	6.51E+07 person-rem
Class 3a	=	4.41E+04 person-rem x 10La = 4.41E+05 person-rem
Class 3b	=	4.41E+04 person-rem x 100La = 4.41E+06 person-rem
Class 4	=	Not analyzed
Class 5	=	Not analyzed
Class 6	=	Not analyzed
Class 7-CFE	=	(4.66E-09 * 1.94E+07 + 6.90E-08 * 6.51E+07) / (4.66E-09 + 6.90E-08) = 6.22E+07 person-rem
Class 7-CFL	=	6.87E+06 person-rem
Class 8-SGTR	=	6.51E+07 person-rem
Class 8-ISLOCA	=	6.51E+07 person-rem

For IP3:

Class 1	=	4.41E+04 person-rem (at 1.0La)
Class 2	=	5.08E+07 person-rem
Class 3a	=	4.41E+04 person-rem x 10La = 4.41E+05 person-rem
Class 3b	=	4.41E+04 person-rem x 100La = 4.41E+06 person-rem
Class 4	=	Not analyzed
Class 5	=	Not analyzed
Class 6	=	Not analyzed
Class 7-CFE	=	(1.17E-07 * 2.00E+07 + 7.14E-08 * 5.08E+07) / (1.17E-07 + 7.14E-08) = 3.17E+07 person-rem
Class 7-CFL	=	6.85E+06 person-rem
Class 8-SGTR	=	5.08E+07 person-rem
Class 8-ISLOCA	=	5.08E+07 person-rem

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [3] for all EPRI classes are provided in Table 5.2-1, which includes the values previously presented in Table 4.2-6a and 4.2-6b as well as the Class 3a, 3b, and 7-CFE population doses calculated above.

TABLE 5.2-1
IP2 AND IP3 POPULATION DOSE
FOR POPULATION WITHIN 50 MILES

ACCIDENT CLASS (CONTAINMENT RELEASE TYPE)	DESCRIPTION	IP2 PERSON-REM (0-50 MILES)	IP3 PERSON-REM (0-50 MILES)
1	Containment Intact	4.41E+04	4.41E+04
2	Large Isolation Failures (Failure to Close)	6.51E+07	5.08E+07
3a	Small Isolation Failures (liner breach)	4.41E+05	4.41E+05
3b	Large Isolation Failures (liner breach)	4.41E+06	4.41E+06
4	Small Isolation Failures (Failure to seal – Type B)	N/A	N/A
5	Small Isolation Failures (Failure to seal - Type C)	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A
7-CFE	Failures Induced by Phenomena (Early)	6.22E+07	3.17E+07
7-CFL	Failures Induced by Phenomena (Late)	6.87E+06	6.85E+06
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	6.51E+07	5.08E+07
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	6.51E+07	5.08E+07

The above population doses, when multiplied by the frequency results presented in Table 5.1-1, yield the IP2 and IP3 baseline mean dose risk for each EPRI accident class. These results are presented in Table 5.2-2a for IP2 and in Table 5.2-2b for IP3. Note that the additional contribution to EPRI Class 3b from the corrosion analysis as described in Section 4.4 is also included in these tables.

TABLE 5.2-2A
IP2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 3 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) [1]
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	Containment Intact [2]	4.41E+04	7.74E-06	3.41E-01	7.74E-06	3.41E-01	-4.14E-06
2	Large Isolation Failures (Failure to Close)	6.51E+07	1.11E-08	7.23E-01	1.11E-08	7.23E-01	--
3a	Small Isolation Failures (liner breach)	4.41E+05	9.73E-08	4.29E-02	9.73E-08	4.29E-02	--
3b	Large Isolation Failures (liner breach)	4.41E+06	2.43E-08	1.07E-01	2.44E-08	1.08E-01	4.14E-4
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

TABLE 5.2-2A
IP2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS;
CHARACTERISTIC OF CONDITIONS FOR 3 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) [1]
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7-CFE	Failures Induced by Phenomena (Early)	6.22E+07	7.37E-08	4.58E+00	7.37E-08	4.58E+00	--
7-CFL	Failures Induced by Phenomena (Late)	6.87E+06	2.71E-06	1.86E+01	2.71E-06	1.86E+01	--
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	6.51E+07	1.05E-06	6.80E+01	1.05E-06	6.80E+01	--
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	6.51E+07	2.77E-08	1.80E+00	2.77E-08	1.80E+00	--
CDF	**All CET end states**		**1.17E-05**	**9.426E+01**	**1.17E-05**	**9.426E+01**	**4.10E-4**

[1] Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years. The additional frequency added to Class 3b is subtracted from Class 1 and the population dose rates are recalculated. This results in a small reduction to the Class 1 dose rate and an increase to the Class 3b dose rate.

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

TABLE 5.2-2B
IP3 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 3 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	Containment Intact (2)	4.41E+04	1.11E-05	4.91E-01	1.11E-05	4.91E-01	-5.32E-6
2	Large Isolation Failures (Failure to Close)	5.08E+07	3.99E-09	2.03E-01	3.99E-09	2.03E-01	--
3a	Small Isolation Failures (liner breach)	4.41E+05	1.25E-07	5.51E-02	1.25E-07	5.51E-02	--
3b	Large Isolation Failures (liner breach)	4.41E+06	3.13E-08	1.38E-01	3.14E-08	1.38E-01	5.32E-4
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

TABLE 5.2-2B
IP3 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 3 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) [1]
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7-CFE	Failures Induced by Phenomena (Early)	3.17E+07	1.88E-07	5.97E+00	1.88E-07	5.97E+00	--
7-CFL	Failures Induced by Phenomena (Late)	6.85E+06	2.17E-06	1.49E+01	2.17E-06	1.49E+01	--
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	5.08E+07	9.77E-07	4.96E+01	9.77E-07	4.96E+01	--
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	5.08E+07	1.93E-07	9.80E+00	1.93E-07	9.80E+00	--
CDF	**All CET end states**		**1.48E-05**	**8.114E+01**	**1.48E-05**	**8.115E+01**	**5.27E-4**

(1) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years. The additional frequency added to Class 3b is subtracted from Class 1 and the population dose rates are recalculated. This results in a small reduction to the Class 1 dose rate and an increase to the Class 3b dose rate.

(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 baseline IP2 and IP3 doses compare reasonably with other plants given the relative population densities surrounding each location:

PLANT	ANNUAL DOSE (PERSON-REM/YR)	REFERENCE
Indian Point 2	94.3	[Table 5.2-2a]
Indian Point 3	81.1	[Table 5.2-2b]
Peach Bottom 2	8.6	[22]
Farley Unit 1, 2	1.5, 2.4	[23]
Crystal River	1.4	[24]

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 year interval).

Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the EPRI 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 5.3-1a for IP2 and in Table 5.3-1b for IP3.

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 not detecting a leak 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 5.3-2a for IP2 and in Table 5.3-2b for IP3.

TABLE 5.3-1A
IP2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 1 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) [(1)]
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	Containment Intact [(2)]	4.41E+04	7.46E-06	3.29E-01	7.45E-06	3.29E-01	-2.38E-05
2	Large Isolation Failures (Failure to Close)	6.51E+07	1.11E-08	7.23E-01	1.11E-08	7.23E-01	--
3a	Small Isolation Failures (liner breach)	4.41E+05	3.24E-07	1.43E-01	3.24E-07	1.43E-01	--
3b	Large Isolation Failures (liner breach)	4.41E+06	8.10E-08	3.57E-01	8.15E-08	3.60E-01	2.38E-3
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

TABLE 5.3-1A
IP2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 1 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7-CFE	Failures Induced by Phenomena (Early)	6.22E+07	7.37E-08	4.58E+00	7.37E-08	4.58E+00	--
7-CFL	Failures Induced by Phenomena (Late)	6.87E+06	2.71E-06	1.86E+01	2.71E-06	1.86E+01	--
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	6.51E+07	1.05E-06	6.80E+01	1.05E-06	6.80E+01	--
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	6.51E+07	2.77E-08	1.80E+00	2.77E-08	1.80E+00	--
CDF	**All CET end states**		**1.17E-05**	**9.460E+01**	**1.17E-05**	**9.460E+01**	**2.35E-3**

(1) Only release classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years. The additional frequency added to Class 3b is subtracted from Class 1 and the population dose rates are recalculated. This results in a small reduction to the Class 1 dose rate and an increase to the Class 3b dose rate.

(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 5.3-1B
IP3 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS;
CHARACTERISTIC OF CONDITIONS FOR 1 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	Containment Intact (2)	4.41E+04	1.08E-05	4.75E-01	1.08E-05	4.75E-01	-3.05E-5
2	Large Isolation Failures (Failure to Close)	5.08E+07	3.99E-09	2.03E-01	3.99E-09	2.03E-01	--
3a	Small Isolation Failures (liner breach)	4.41E+05	4.16E-07	1.84E-01	4.16E-07	1.84E-01	--
3b	Large Isolation Failures (liner breach)	4.41E+06	1.04E-07	4.59E-01	1.05E-07	4.62E-01	3.05E-3
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

TABLE 5.3-1B
IP3 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 1 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7-CFE	Failures Induced by Phenomena (Early)	3.17E+07	1.88E-07	5.97E+00	1.88E-07	5.97E+00	--
7-CFL	Failures Induced by Phenomena (Late)	6.85E+06	2.17E-06	1.49E+01	2.17E-06	1.49E+01	--
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	5.08E+07	9.77E-07	4.96E+01	9.77E-07	4.96E+01	--
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	5.08E+07	1.93E-07	9.80E+00	1.93E-07	9.80E+00	--
CDF	**All CET end states**		**1.48E-05**	**8.158E+01**	**1.48E-05**	**8.158E+01**	**3.02E-3**

(1) Only release classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years. The additional frequency added to Class 3b is subtracted from Class 1 and the population dose rates are recalculated. This results in a small reduction to the Class 1 dose rate and an increase to the Class 3b dose rate.

(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 5.3-2A
IP2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 1 IN 15 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	Containment Intact (2)	4.41E+04	7.25E-06	3.20E-01	7.25E-06	3.20E-01	-5.51E-05
2	Large Isolation Failures (Failure to Close)	6.51E+07	1.11E-08	7.23E-01	1.11E-08	7.23E-01	--
3a	Small Isolation Failures (liner breach)	4.41E+05	4.86E-07	2.15E-01	4.86E-07	2.15E-01	--
3b	Large Isolation Failures (liner breach)	4.41E+06	1.22E-07	5.36E-01	1.23E-07	5.42E-01	5.51E-3
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

TABLE 5.3-2A
IP2 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS;
CHARACTERISTIC OF CONDITIONS FOR 1 IN 15 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7-CFE	Failures Induced by Phenomena (Early)	6.22E+07	7.37E-08	4.58E+00	7.37E-08	4.58E+00	--
7-CFL	Failures Induced by Phenomena (Late)	6.87E+06	2.71E-06	1.86E+01	2.71E-06	1.86E+01	--
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	6.51E+07	1.05E-06	6.80E+01	1.05E-06	6.80E+01	--
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	6.51E+07	2.77E-08	1.80E+00	2.77E-08	1.80E+00	--
CDF	**All CET end states**		**1.17E-05**	**9.484E+01**	**1.17E-05**	**9.484E+01**	**5.46E-3**

(1) Only release classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years. The additional frequency added to Class 3b is subtracted from Class 1 and the population dose rates are recalculated. This results in a small reduction to the Class 1 dose rate and an increase to the Class 3b dose rate.

(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 5.3-2B
IP3 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS;
CHARACTERISTIC OF CONDITIONS FOR 1 IN 15 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	Containment Intact (2)	4.41E+04	1.05E-05	4.64E-01	1.05E-05	4.64E-01	-7.08E-5
2	Large Isolation Failures (Failure to Close)	5.08E+07	3.99E-09	2.03E-01	3.99E-09	2.03E-01	--
3a	Small Isolation Failures (liner breach)	4.41E+05	6.25E-07	2.76E-01	6.25E-07	2.76E-01	--
3b	Large Isolation Failures (liner breach)	4.41E+06	1.56E-07	6.89E-01	1.58E-07	6.96E-01	7.08E-3
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A	N/A	N/A	N/A

TABLE 5.3-2B
IP3 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 1 IN 15 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
7-CFE	Failures Induced by Phenomena (Early)	3.17E+07	1.88E-07	5.97E+00	1.88E-07	5.97E+00	--
7-CFL	Failures Induced by Phenomena (Late)	6.85E+06	2.17E-06	1.49E+01	2.17E-06	1.49E+01	--
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	5.08E+07	9.77E-07	4.96E+01	9.77E-07	4.96E+01	--
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	5.08E+07	1.93E-07	9.80E+00	1.93E-07	9.80E+00	--
CDF	**All CET end states**		**1.48E-05**	**8.189E+01**	**1.48E-05**	**8.190E+01**	**7.01E-3**

(1) Only release classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years. The additional frequency added to Class 3b is subtracted from Class 1 and the population dose rates are recalculated. This results in a small reduction to the Class 1 dose rate and an increase to the Class 3b dose rate.

(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-06/yr and increases in LERF below 1E-07/yr, and small changes in LERF as below 1E-06/yr. Because the ILRT does not impact CDF for IP2 and IP3, the relevant metric is LERF.

For IP2 and IP3, 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 EPRI guidance methodology and the NRC SE). Based on the original 3-in-10 year test interval assessment from Tables 5.2-2a and 5.2-2b, the Class 3b frequency is 2.44E-08/yr for IP2 and 3.14E-08/yr for IP3, which includes the corrosion effect of the containment liner. Based on a ten-year test interval from Tables 5.3-1a and 5.3-1b, the Class 3b frequency is 8.15E-08/yr for IP2 and 1.05E-07/yr for IP3; and, based on a fifteen-year test interval from Tables 5.3-2a and 5.3-2b, it is 1.23E-07/yr for IP2 and 1.58E-07/yr for IP3. 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 9.84E-08/yr for IP2 and 1.26E-07/yr for IP3. Similarly, the increase in LERF due to increasing the interval from 10 to 15 years (including corrosion effects) is 4.13E-08/yr for IP2 and 5.31E-08/yr for IP3. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF is well within Region II of Figure 4 of Reference [4] (i.e., the acceptance criteria for small changes in LERF) when comparing the 15 year results to the original 3-in-10 year requirement.

5.5 STEP 5 - DETERMINE THE IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY

Another parameter that 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 and, consistent with the EPRI guidance, the small isolation failures (Class 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the EPRI methodology [3]. The NRC SE has noted a change in CCFP of <1.5% as the acceptance criterion to be used as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. Table 5.5-1 shows the CCFP values that result from the assessment for the various testing intervals including corrosion effects in which the flaw rate is assumed to double every five years.

TABLE 5.5-1
IP2 AND IP3 ILRT CONDITIONAL CONTAINMENT FAILURE PROBABILITIES

UNIT	CCFP 3 IN 10 YRS	CCFP 1 IN 10 YRS	CCFP 1 IN 15 YRS	$\Delta CCFP_{15-3}$	$\Delta CCFP_{15-10}$
Indian Point 2	33.19%	33.67%	34.03%	0.84%	0.35%
Indian Point 3	24.03%	24.52%	24.88%	0.85%	0.36%

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

The change in CCFP of less than 1% as a result of extending the test interval to 15 years from the original 3-in-10 year requirement is judged to be relatively insignificant, and is less than the NRC SE acceptance criteria of < 1.5%.

5.6 SUMMARY OF INTERNAL EVENTS RESULTS

Table 5.6-1a summarizes the internal events results of this ILRT extension risk assessment for IP2. Table 5.6-1b summarizes the internal events results of this ILRT extension risk assessment for IP3. The results between the 3-in-10 year interval and the 15 year interval compared to the acceptance criteria are then shown in Table 5.6-2 for IP2 and IP3, and it is demonstrated that the acceptance criteria are met.

TABLE 5.6-1A
IP2 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 (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	4.41E+04	7.74E-06	3.41E-01	7.45E-06	3.29E-01	7.25E-06	3.20E-01
2	6.51E+07	1.11E-08	7.23E-01	1.11E-08	7.23E-01	1.11E-08	7.23E-01
3a	4.41E+05	9.73E-08	4.29E-02	3.24E-07	1.43E-01	4.86E-07	2.15E-01
3b	4.41E+06	2.44E-08	1.08E-01	8.15E-08	3.60E-01	1.23E-07	5.42E-01
7-CFE	6.22E+07	7.37E-08	4.58E+00	7.37E-08	4.58E+00	7.37E-08	4.58E+00
7-CFL	6.87E+06	2.71E-06	1.86E+01	2.71E-06	1.86E+01	2.71E-06	1.86E+01
8-SGTR	6.51E+07	1.05E-06	6.80E+01	1.05E-06	6.80E+01	1.05E-06	6.80E+01
8-ISLOCA	6.51E+07	2.77E-08	1.80E+00	2.77E-08	1.80E+00	2.77E-08	1.80E+00
Total		1.17E-05	9.426E+01	1.17E-05	9.460E+01	1.17E-05	9.484E+01
ILRT Dose Rate (person-rem/yr) from 3a and 3b		1.51E-01		5.02E-01		7.56E-01	
Delta Total Dose Rate[1]	From 3 yr	---		3.39E-01		5.84E-01	
	From 10 yr	---		---		2.45E-01	
3b Frequency (LERF)		2.44E-08		8.15E-08		1.23E-07	
Delta 3b LERF	From 3 yr	---		5.71E-08		9.84E-08	
	From 10 yr	---		---		4.13E-08	
CCFP %		33.19%		33.67%		34.03%	
Delta CCFP %	From 3 yr	---		0.49%		0.84%	
	From 10 yr	---		---		0.35%	

(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 due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

TABLE 5.6-1B
IP3 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 (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	4.41E+04	1.11E-05	4.91E-01	1.08E-05	4.75E-01	1.05E-05	4.64E-01
2	5.08E+07	3.99E-09	2.03E-01	3.99E-09	2.03E-01	3.99E-09	2.03E-01
3a	4.41E+05	1.25E-07	5.51E-02	4.16E-07	1.84E-01	6.25E-07	2.76E-01
3b	4.41E+06	3.14E-08	1.38E-01	1.05E-07	4.62E-01	1.58E-07	6.96E-01
7-CFE	3.17E+07	1.88E-07	5.97E+00	1.88E-07	5.97E+00	1.88E-07	5.97E+00
7-CFL	6.85E+06	2.17E-06	1.49E+01	2.17E-06	1.49E+01	2.17E-06	1.49E+01
8-SGTR	5.08E+07	9.77E-07	4.96E+01	9.77E-07	4.96E+01	9.77E-07	4.96E+01
8-ISLOCA	5.08E+07	1.93E-07	9.80E+00	1.93E-07	9.80E+00	1.93E-07	9.80E+00
Total		1.48E-05	8.115E+01	1.48E-05	8.158E+01	1.48E-05	8.190E+01
ILRT Dose Rate (person-rem/yr) from 3a and 3b		1.93E-01		6.46E-01		9.72E-01	
Delta Total Dose Rate(1)	From 3 yr	---		4.36E-01		7.51E-01	
	From 10 yr	---		---		3.15E-01	
3b Frequency (LERF)		3.14E-08		1.05E-07		1.58E-07	
Delta 3b LERF	From 3 yr	---		7.34E-08		1.26E-07	
	From 10 yr	---		---		5.31E-08	
CCFP %		24.03%		24.52%		24.88%	
Delta CCFP %	From 3 yr	---		0.49%		0.85%	
	From 10 yr	---		---		0.36%	

(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 due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

TABLE 5.6-2
IP2 AND IP3 ILRT EXTENSION COMPARISON TO ACCEPTANCE CRITERIA

Unit	ΔLERF	ΔPerson-rem/yr	ΔCCFP
Indian Point 2	9.84E-8/yr	0.584/yr (0.62%)	0.84%
Indian Point 3	1.26E-7/yr	0.751/yr (0.93%)	0.85%
Acceptance Criteria	**<1.0E-6/yr**	**<1.0 person-rem/yr or <1.0%**	**<1.5%**

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 method chosen to account for external events contributions is similar to that used in the SAMA analysis [20] in which a multiplier was applied to the internal events results based on information from the IPEEE [8, 9]. Similar to that provided in the SAMA analysis, a description of the external events contribution to risk at IP2 and IP3 is provided below.

5.7.1 Indian Point 2 External Events Discussion

The IP2 Individual Plant Examination of External Events (IPEEE) included quantitative CDF results for high winds, seismic, and fire contributors. Each of these is discussed below.

A high wind analysis was performed for the IP2 IPEEE. Conservative assumptions in the high wind PRA analysis included the following.

- Offsite power was assumed to be lost for all high wind events.
- Building frame failures were assumed to cause failure of all equipment within the building.
- Missile (high wind projectile) impact on a structure was assumed to cause failure of all equipment within that structure.
- Likelihood of missile (high wind projectile) strikes was assumed to be independent of the intensity of the hazard.
- Both onsite and offsite alternate power sources (gas turbines) were assumed to fail given failure of a more robust structure.

The core damage frequency contribution associated with high wind events was estimated to be 3.03E-05/yr. As described above, this is a conservative value. In addition, plant changes, improved equipment performance data, and modeling improvements since the issuance of the IP2 IPEEE have demonstrated that the response of plant systems as modeled at that time was conservative. This can be seen from the reduction in internal events CDF from 2.85E-05/yr at the time the IPEEE was developed to the present value of 1.17E-05/yr. Although conservative, consistent with the SAMA analysis, the wind risk contribution of 3.03E-05/yr is maintained to determine the potential external events impact in the ILRT extension assessment.

A seismic PRA analysis was performed for the seismic portion of the IP2 IPEEE. The seismic PRA analysis was a conservative analysis. Therefore, its results should not be compared directly with the best-estimate internal events results. Conservative assumptions in the seismic PRA analysis included the following.

- Sequences in the seismic PRA involving loss of off-site power were assumed to be unrecoverable. If off-site power was recovered following a seismic event, there would be many more systems available to maintain core cooling and containment integrity than were credited for those sequences.
- A single, conservative, surrogate element whose failure leads directly to core damage was used in the seismic risk quantification to model the most seismically rugged components.
- Seismic-induced ATWS was considered in the analysis, but no credit was included for manual scram or mitigation of ATWS using the boration system. This conservatively resulted in most seismic-induced ATWS events leading to consequential core damage.
- Redundant components were conservatively assumed to be completely correlated by treating them as if they were one component for the purpose of determining the probability of seismic induced failures.
- Several systems were assumed to be unavailable during a seismic event, including:
    a. the city water system, which can be used to supply backup cooling to the charging pumps if CCW is lost, as an alternate source of suction to the AFW pumps and to provide alternate cooling to the RHR and SI pumps;
    b. the primary water system, which can also be used as a backup to CCW to supply cooling to the RHR and SI pumps; and
    c. the onsite and offsite gas turbine generators, which can provide alternate station power.
- No credit was taken for recovery of power through the alternate safe shutdown system (ASSS).

The seismic CDF in the IPEEE was originally estimated to be 1.46E-05/yr. As a result of an IPEEE recommendation, the CCW surge tank hold-down bolts were upgraded, reducing the seismic CDF to 1.06E-05/yr. Although it remains conservative, consistent with the SAMA analysis, the seismic risk contribution of 1.06E-05/yr is maintained to determine the potential external events impact in the ILRT extension assessment.

The conservative EPRI FIVE methodology was used for initial screening of fire zones in the IP2 IPEEE fire analysis. Unscreened fire zones were then analyzed in more detail using a fire PRA approach. The sum of the resulting fire zone CDF values is approximately 1.84E-05/yr. Conservative assumptions in the IP2 IPEEE fire analysis include the following.

- The frequency and severity of fires were generally conservatively overestimated in the generic IPEEE fire analysis methods. A revised NRC fire events database indicates a trend toward lower frequency and less severe fires. This trend reflects improved housekeeping, reduction in transient fire hazards, and other improved fire protection steps at utilities.
- Cable failure due to fire damage was assumed to arise from open circuits, hot short circuits, and short circuits to ground. In damaging a cable, the analysis addressed the ability of the fire to induce the conductor failure mode of concern. Hot shorts were conservatively assigned a probability of 0.1, which was applied to all single phase, AC control circuit or DC power and control circuit cases regardless of whether the wires were in the same multi-conductor.
- A plant trip was assumed for all fires, including those for which immediate operator actions are not specified in emergency response procedures.
- PORV block valves were assumed to be in the more limiting position (open or closed) to maximize the impact of the fire.
- The main feedwater and condensate systems were assumed to be unavailable in all scenarios, even when their power source was not impacted by the fire scenario. Use of these systems for recovery, following a failure of AFW, is addressed in current plant procedures.
- All sequences involving induced RCP seal LOCAs were assumed to lead to complete seal failure. Although casualty cables exist for powering ECCS pumps from the ASSS power source, the ASSS was assumed to be ineffective in mitigating induced LOCAs.
- The currently accepted RCP seal LOCA methodology is more detailed and provides sequences with varying leakage rates. Under that current methodology, a majority of seal LOCAs remain within the capability of a charging pump (which has hardwired ASSS transfer capability) to provide makeup.

As noted previously, plant changes, improved equipment performance data and modeling improvements since the issuance of the IP2 IPEEE have demonstrated that the response of plant systems as modeled at that time was conservative. This can be seen from the reduction in internal events CDF from 2.85E-05/yr at the time the IPEEE was developed to the present value of 1.17E-5/yr., a reduction factor of 2.4. Factoring in the additional conservatisms in the fire analysis noted above, an overall reduction factor of 2 is reasonable which is consistent with the assumption used in the SAMA analysis [20]. The IPEEE fire CDF value, reduced by a factor of two, is 9.20E-06/yr.

The IP2 Individual Plant Examination of External Events (IPEEE) concluded for "Other" external events, with the exception of "high wind" events as noted above, that no undue risks are present that might contribute to CDF with a predicted frequency in excess of 1.0E-06/yr. As these events are not dominant contributors to external event risk and quantitative analysis of these events is not practical, they are considered negligible in estimation of the external events impact on the ILRT extension assessment.

In summary, the combination of the IPEEE high wind CDF and the reduced seismic and fire CDF values described above results in an external events risk estimate of 5.01E-05/yr, which is 4.3 times higher than the internal events CDF (1.17E-05/yr).

5.7.2 Indian Point 3 External Events Discussion

The IP3 Individual Plant Examination of External Events (IPEEE) concluded for high winds, floods, and "Other" external events that no undue risks are present that might contribute to CDF with a predicted frequency in excess of 1.0E-06/yr. Note that at IP3 (compared to IP2), the EDGs are in separate concrete bunkered cells and as such are not susceptible to high winds. In any event, as these other events are not dominant contributors to external event risk and quantitative analysis of these events is not practical, they are considered negligible in estimation of the external events impact on the ILRT extension assessment. The IPEEE analyses using the seismic PRA and fire PRA provided quantitative, but conservative, results. Therefore, the results were combined as described below to represent the total external events risk.

A seismic PRA analysis was performed for the seismic portion of the IP3 IPEEE. The seismic PRA analysis is a conservative analysis. Therefore, its results should not be compared directly with the best-estimate internal events results. Conservative assumptions in the seismic PRA analysis included the following.

- Each of the sequences in the seismic PRA assumes unrecoverable loss of off-site power. If off-site power was maintained, or recovered, following a seismic event, there would be many more systems available to maintain core cooling and containment integrity than were credited in the analysis.
- Seismic events were assumed to induce a small loss of coolant accident (LOCA) in addition to a loss of offsite power.
- A single, conservative, surrogate element whose failure leads directly to core damage was used in the seismic risk quantification to model the most seismically rugged components.
- Redundant components were conservatively assumed to be completely correlated by treating them as if they were one component for the purpose of determining the probability of seismic induced failures.

- The ATWS event tree was conservatively simplified so that all conditions which lead to a failure to trip result in core damage, without the benefit of emergency boration or other mitigating systems.
- Because there is little industry experience with crew actions following seismic events, human actions were conservatively characterized.

The seismic CDF in the IPEEE was conservatively estimated to be 4.40E-05/yr. As described above, this is a conservative value. The seismic PRA CDF has been re-evaluated to reflect updated random component failure probabilities and to model recovery of onsite power and local operation of the turbine-driven AFW pump. The updated seismic CDF is 2.65E-05/yr. Although it remains conservative, consistent with the SAMA analysis, the seismic risk contribution of 2.65E-05/yr is maintained to determine the external events impact on the ILRT extension assessment.

The EPRI Fire PRA Implementation Guide was followed for the IP3 IPEEE fire analysis. The EPRI Fire Induced Vulnerability Evaluation (FIVE) method was used for the initial screening, for treatment of transient combustibles, and as the source of fire frequency data. The sum of the resulting fire zone CDF values is approximately 5.58E-05/yr. Conservatisms in the IP3 IPEEE fire analysis include the following.

- The frequency and severity of fires were generally conservatively overestimated. A revised NRC fire events database indicates a trend toward lower frequency and less severe fires. This trend reflects improved housekeeping, reduction in transient fire hazards, and other improved fire protection steps at utilities.
- There is little industry experience with crew actions following fires. This led to conservative characterization of crew actions in the IPEEE fire analysis. Because CDF is strongly correlated with crew actions, this conservatism has a profound effect on fire results.
- Hot gas layer temperature timing calculations were based on simplified analyses (versus more detailed calculations such as GOTHIC or even COMPBURN) which are believed to result in more severe timing (i.e., shorter time to equipment failure).
- Heat and combustion products from a fire within a zone were assumed to be confined within the zone. Heat loss through separating zones was not considered; nor was heat loss through open equipment hatches, ladder ways, open doorways, or unsealed penetrations.
- Cable failure due to fire damage was assumed to arise from open circuits, hot shorts circuits, and short circuits to ground. In damaging a cable, the fire was always assumed to induce the conductor failure mode of concern.
- A plant trip was assumed for all fires, including those for which immediate operator actions are not specified in emergency response procedures.
- For several fire zones, a minimum heat requirement for target damage was estimated.
- Propagation of fires in cable spreading room trays and electrical tunnels was modeled using a maximum heat release rate. This results in a shorter time to damage than

the five-minute delay using heat release rate scaling factors as a function of distance recommended in the EPRI fire PRA implementation guide.

Implementation of the IP3 IPEEE recommendations reduced the fire risk. The fire suppression system in the 480V switchgear room was restored to automatic actuation, and realignment and rerouting of the power feeds to the EDG exhaust fans and engine auxiliaries in emergency diesel generator room 31, emergency diesel generator room 32, and emergency diesel generator room 33 significantly reduce the respective fire zone's CDF. In addition, restoration of the 480V switchgear room fire suppression system to automatic actuation results in a similar reduction in the fire zone 14/37A multiple compartment fire CDF. Consequently, the IPEEE fire CDF value was reduced from 5.58E-05/yr to 2.55E-05/yr. Although it remains conservative, consistent with the SAMA analysis, the fire risk contribution of 2.55E-05/yr is maintained to determine the potential external event impact on the ILRT extension assessment.

In summary, combining the reduced seismic and fire CDF values results in an external events risk estimate of 5.20E-05/yr, which is 3.5 times higher than the internal events CDF (1.48E-05/yr).

5.7.3 Additional Seismic Risk Discussion

As an additional consideration, it can be noted that in June 2013, Entergy submitted information to the NRC that addressed some conservatisms in the original IPEEE analyses, and indicated that the seismic CDF risk at IP2 and IP3 are both actually less than 1.0E-05/yr [25]. However, to maintain consistency with the approach utilized in the SAMA analysis, the additional information will not be factored into this analysis but is noted here for completeness.

5.7.4 External Events Impact Summary

Table 5.7-1 summarizes the external events CDF contribution for IP2 and IP3. Although noted as conservative, these values are consistent with that used in the SAMA analysis [20].

TABLE 5.7-1
EXTERNAL EVENTS CONTRIBUTOR SUMMARY [20]

EXTERNAL EVENT INITIATOR GROUP	IP2 CDF (1/YR)	IP3 CDF (1/YR)
Seismic	1.06E-05	2.65E-05
Internal Fire	9.20E-06	2.55E-05
High Winds	3.03E-05	Screened
Other Hazards	Screened	Screened
Total (for initiators with CDF available)	**5.01E-05**	**5.20E-05**
Internal Events CDF	1.17E-05	1.48E-05
External Events Multiplier	**4.28**	**3.51**

From Table 5.7-1, the external events multiplier for IP2 is conservatively estimated to be 4.28 and for IP3, it is conservatively estimated to be 3.51.

5.7.5 External Events Impact on ILRT Extension Assessment

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 5.6-1a for IP2 as 2.44E-08/yr, 8.15E-08/yr, and 1.23E-07/yr, respectively. Using an external events multiplier of 4.28 for IP2, 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 hazards risk, is estimated as shown in Table 5.7-2a. Similarly, the EPRI Class 3b frequencies shown in Table 5.6-1b for IP3 are 3.14E-08/yr, 1.05E-07/yr, and 1.58E-07/yr. Using an external events multiplier of 3.51 for IP3, 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 hazards risk, is estimated as shown in Table 5.7-2b.

TABLE 5.7-2A
IP2 3B (LERF/YR) 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	2.44E-08	8.15E-08	1.23E-07	9.84E-08
External Events Contribution (Internal Events CDF x 4.28)	1.05E-07	3.49E-07	5.26E-07	4.22E-07
Combined (Internal + External)	1.29E-07	4.31E-07	6.49E-7	5.20E-07

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

Thus for IP2, the total increase in LERF (measured from the baseline 3-per-10 year ILRT interval to the proposed 1-per-15 year frequency) due to the combined internal and external events contribution is estimated as 5.20E-07/yr, which includes the age adjusted steel liner corrosion likelihood.

TABLE 5.7-2B
IP3 3B (LERF/YR) 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	3.14E-08	1.05E-07	1.58E-07	1.26E-07
External Events Contribution (Internal Events CDF x 3.51)	1.10E-07	3.67E-07	5.53E-07	4.43E-07
Combined (Internal + External)	1.41E-07	4.72E-07	7.11E-7	5.70E-07

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

Thus for IP3, the total increase in LERF (measured from the baseline 3-per-10 year ILRT interval to the proposed 1-per-15 year frequency) due to the combined internal and external events contribution is estimated as 5.70E-07/yr, which includes the age adjusted steel liner corrosion likelihood.

The other acceptance criteria for the ILRT extension risk assessment can be similarly derived using the multiplier approach. The results between the 3-in-10 year interval and the 15 year interval compared to the acceptance criteria are shown in Table 5.7-3. As can be seen, the impact from including the external events contributors would not change the conclusion of the risk assessment. That is, the acceptance criteria are all met such that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years has been demonstrated to be small. Note that a bounding analysis for the total LERF contribution follows Table 5.7-3 to demonstrate that the total LERF value for IP2 and IP3 is less than 1.0E-5/yr consistent with the requirements for a "Small Change" in risk of the RG 1.174 acceptance guidelines.

TABLE 5.7-3
COMPARISON TO ACCEPTANCE CRITERIA INCLUDING EXTERNAL EVENTS CONTRIBUTION FOR IP2 AND IP3

Contributor	ΔLERF	ΔPerson-rem/yr	ΔCCFP
IP2 Internal Events	9.84E-8/yr	0.584/yr (0.62%)	0.84%
IP2 External Events	4.22E-7/yr	2.50/yr (0.62%)	0.84%
Indian Point 2 Total	**5.20E-7/yr**	**3.09/yr (0.62%)**	**0.84%**
IP3 Internal Events	1.26E-7/yr	0.751/yr (0.93%)	0.85%
IP3 External Events	4.43E-7/yr	2.63/yr (0.93%)	0.85%
Indian Point 3 Total	**5.70E-7/yr**	**3.38/yr (0.93%)**	**0.85%**
Acceptance Criteria	**<1.0E-6/yr**	**<1.0 person-rem/yr or <1.0%**	**<1.5%**

The 5.20E-07/yr increase in LERF for IP2 and the 5.70E-07/yr increase in LERF for IP3 due to the combined internal and external events from extending the ILRT frequency from 3-per-10 years to 1-per-15 years falls within Region II between 1.0E-7 to 1.0E-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 1.0E-5/yr. Similar bounding assumptions regarding the external event contributions that were made above are used for the total LERF estimate.

From Table 4.2-1, the total LERF due to postulated internal event accidents is 1.16E-06/yr for IP2 and 1.25E-06/yr for IP3. Although some of the LERF contributors may not be applicable to external events initiators, the base LERF distribution due to external events is assumed to be the same as the internal events contribution. The total LERF values for IP2 and IP3 are then shown in Table 5.7-4.

TABLE 5.7-4
IMPACT OF 15-YR ILRT EXTENSION ON LERF FOR IP2 AND IP3

LERF CONTRIBUTOR	IP2 (1/YR)	IP3 (1/YR)
Internal Events LERF	1.16E-06	1.25E-06
External Events LERF	4.97E-06 [Internal Events LERF * 4.28]	4.38E-06 [Internal Events LERF * 3.51]
Internal Events LERF due to ILRT (at 15 years) [(1)]	1.23E-07	1.58E-07
External Events LERF due to ILRT (at 15 years) [(1)]	5.26E-07	5.53E-07
Total	**6.78E-06/yr**	**6.34E-06/yr**

(1) Including age adjusted steel liner corrosion likelihood as reported in Table 5.7-2a for IP2 and Table 5.7-2b for IP3.

As can be seen, the estimated upper bound LERF for IP2 is estimated as 6.78E-06/yr and for IP3 it is 6.34E-06/yr. These values are both less than the RG 1.174 requirement to demonstrate that the total LERF due to internal and external events is less than 1.0E-5/yr.

5.7.6 <u>Alternative Approach for External Events Impact on ILRT Extension Assessment</u>

The approach above described in Section 5.7.5 for the external events impact is consistent with that used in the Palisades ILRT extension risk assessment evaluation that was submitted by Entergy [26] and approved by the NRC [27]. As shown, the IP2 and IP3 results fall within the value in the NRC SER for a small increase in population dose, as defined by percent increase in dose (i.e., <1.0% person-rem/yr). However, since the IP2 and IP3 results rely on that criterion rather than the absolute increase in dose criteria (i.e., < 1.0 person-rem/yr), additional information is provided to further demonstrate that the percent increase in dose criteria is not exceeded.

To do this, a reasonable estimate for the base case dose risk associated with external events must be determined. In this case, each EPRI accident class is re-examined considering the potential contribution for external events. Since the Class 1 frequency is determined based on remaining contribution not assigned to other classes, the discussion appears in reverse order starting with EPRI Class 8 and ending with EPRI Class 1. However, EPRI Class 2 is discussed prior to Class 3 since its value is used in the final determination of the Class 3 frequencies.

Class 8 Sequences

This group represents sequences where containment bypass occurs (SGTR or ISLOCA). ISLOCA and SGTR initiators are deemed inapplicable to the external events assessment so only induced SGTR scenarios need to be considered. From the frequency information provided in Table 4.2-1 for IP2, the induced SGTR contribution to core damage is about 0.75% and for IP3 it represented about 0.39%. A value of 0.5% is assumed for the external events contribution for both IP2 and IP3. A High Early release magnitude dose is assigned.

For IP2:

Class_8 = 0.005 * [IP2 External Events CDF]
= 0.005 * [5.01E-05]
= 2.51E-07/yr

For IP3:

Class_8 = 0.005 * [IP3 External Events CDF]
= 0.005 * [5.20E-05]
= 2.60E-07/yr

Class 7 Sequences

This group represents containment failure induced by early and late severe accident phenomena. From Table 5.1-1 for IP2, the contribution from the early Class 7 sequences is about 0.6% and for IP3 it represented about 1.3%. A value of 1.0% is assumed for the external events contribution for both IP2 and IP3. A High Early release magnitude dose is assigned. From Table 5.1-1 for IP2, the contribution from the late Class 7 sequences is about 23% and for IP3 it represented about 15%. However, since the external events contributors are more dominated by unrecoverable SBO-like scenarios, a value of 50% is assumed for the external events contribution for both IP2 and IP3. A High Late release magnitude dose is assigned.

For IP2:

Class_7-CFE = 0.01 * [IP2 External Events CDF]
= 0.01 * [5.01E-05]
= 5.01E-07/yr

Class_7-CFL = 0.50 * [IP2 External Events CDF]
= 0.50 * [5.01E-05]
= 2.51E-05/yr

For IP3:

Class_7-CFE = 0.01 * [IP3 External Events CDF]
= 0.01 * [5.20E-05]
= 5.20E-07/yr

Class_7-CFL = 0.50 * [IP3 External Events CDF]
= 0.50 * [5.20E-05]
= 2.60E-05/yr

<u>Class 4, 5, and 6 Sequences</u>

Similar to the internal events assessment, because these failures are unaffected by the Type A ILRT, these groups are not evaluated any further in this analysis.

<u>Class 2 Sequences</u>

This group consists of large containment isolation failures. From the frequency information provided in Table 4.2-1 for IP2, the internal events contribution to this accident class was approximately 0.1% of the CDF and for IP3 it represented about 0.03%. Since seismic and fire initiated events would likely be more susceptible to this failure mode, the larger contribution of 0.1% is assumed for both IP2 and IP3. The population doses are assigned the same as the Class 2 scenarios in the internal events assessment.

For IP2:

Class_2 = 0.001 * [IP2 External Events CDF]
= 0.001 * [5.01E-05]
= 5.01E-08/yr

For IP3:

Class_2 = 0.001 * [IP3 External Events CDF]
= 0.001 * [5.20E-05]
= 5.20E-08/yr

<u>Class 3 Sequences</u>

Similar to the internal events assessment, the respective frequencies per year are determined as follows:

$PROB_{Class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.0092 (see Section 4.3)

$PROB_{Class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0023 (see Section 4.3)

As described in Section 4.3, additional consideration is made to not apply these failure probabilities to those cases that are already considered LERF scenarios (i.e., the Class 2, Class 7, and Class 8 LERF contributions). This adjustment is made for based on the frequency information described above for IP2 and IP3, respectively as shown below.

For IP2:

Class_3a = 0.0092 * [CDF – (Class 2 + Class 7-CFE + Class 8)]
= 0.0092 * [5.01E-05 – (5.01E-08 + 5.01E-07 + 2.51E-07)]
= 4.54E-07/yr

Class_3b = 0.0023 * [CDF – (Class 2 + Class 7-CFE + Class 8)]
= 0.0023 * [5.01E-05 – (5.01E-08 + 5.01E-07 + 2.51E-07)]
= 1.13E-07/yr

For IP3:

Class_3a = 0.0092 * [CDF – (Class 2 + Class 7-CFE + Class 8)]
= 0.0092 * [5.20E-05 – (5.20E-08 + 5.20E-07 + 2.60E-07)]
= 4.71E-07/yr

Class_3b = 0.0023 * [CDF – (Class 2 + Class 7-CFE + Class 8)]
= 0.0023 * [5.20E-05 – (5.20E-08 + 5.20E-07 + 2.60E-07)]
= 1.18E-07/yr

For this analysis, the associated containment leakage for Class 3a is 10La and 100La for Class 3b, which is consistent with the latest EPRI methodology [3] and the NRC SE [7].

Class 1 Sequences

Similar to the internal events assessment, the frequency is determined by subtracting all containment failure end states including the EPRI/NEI Class 3a and 3b frequency calculated below, from the total CDF. The internal events intact containment dose of 4.41E+04 person-rem for IP2 and IP3 is also utilized.

Summary of Alternative External Events Base Case Dose Assessment

In summary, the accident sequence frequencies that can lead to release of radionuclides to the public have been derived in a manner consistent with the definition of accident classes defined in EPRI 1018243 [3]. These frequencies have been combined with reasonable assumptions regarding the population dose associated with each class to determine the base case population dose risk for external events. This information is provided in Table 5.7-5a for IP2 and in Table 5.7-5b for IP3. Additionally, following the same EPRI methodology utilized for internal events to determine the risk impact assessment of extending the ILRT interval, the external events accident class frequencies indicative of a 15 year ILRT interval are provided in Table 5.7-6a for IP2 and in Table 5.7-6b for IP3.

Table 5.7-7 then shows the changes due to the ILRT extension from 3 year to a 15 year interval in the LERF, person-rem/yr, and CCFP figures of merit. When these values are added to the internal events results, the acceptance criteria are all still met by using this detailed alternative external events evaluation instead of the simple evaluation that was utilized in Section 5.7.5. A comparison to the acceptance criteria is also shown in Table 5.7-7. Note that the ΔLERF, person-rem/yr, and change in CCFCP shown in Table 5.7-7 are all slightly higher than the corresponding values shown in Table 5.7-3. This is because the simple method in Table 5.7-3 assumes the same distribution of LERF contributors exists between the internal and external events models whereas the alternative assessment re-apportions the base case LERF contributions based on more realistic assumptions while conservatively maintaining the total CDF value. That is, since the contribution from SGTR initiators and ISLOCA initiators (which contribute to the base LERF value) are not applicable to the external events contribution, more of the remaining CDF distribution is potentially affected by the ILRT extension as represented by the Class 3b multiplier on CDF (that is not already LERF). Additionally, the alternative detailed assessment leads to slightly different percent increases in person-rem/yr which are a function of the base case dose estimates.

TABLE 5.7-5A
POPULATION DOSE RISK AS A FUNCTION OF ACCIDENT CLASS
(IP2 ALTERNATIVE EXTERNAL EVENTS BASE CASE)

ACCIDENT CLASS (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (1/YR)	DOSE (PERSON-REM)	DOSE RISK (PERSON-REM/YR)
1	Containment Intact	2.37E-05	4.41E+04	1.04E+00
2	Large Isolation Failures (Failure to Close)	5.01E-08	6.51E+07	3.26E+00
3a	Small Isolation Failures (liner breach)	4.54E-07	4.41E+05	2.00E-01
3b	Large Isolation Failures (liner breach)	1.13E-07	4.41E+06	5.00E-01
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A
7-CFE	Failures Induced by Phenomena (Early)	5.01E-07	6.51E+07	3.26E+01
7-CFL	Failures Induced by Phenomena (Late)	2.51E-05	1.63E+07	4.08E+02
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	2.51E-07	6.51E+07	1.63E+01
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	0.00E+00	6.51E+07	0.00E+00
CDF	**All CET End States (Including Intact Case)**	**5.01E-05**		**462.2**

TABLE 5.7-5B
POPULATION DOSE RISK AS A FUNCTION OF ACCIDENT CLASS
(IP3 ALTERNATIVE EXTERNAL EVENTS BASE CASE)

ACCIDENT CLASS (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (1/YR)	DOSE (PERSON-REM)	DOSE RISK (PERSON-REM/YR)
1	Containment Intact	2.46E-05	4.41E+04	1.08E+00
2	Large Isolation Failures (Failure to Close)	5.20E-08	5.08E+07	2.64E+00
3a	Small Isolation Failures (liner breach)	4.71E-07	4.41E+05	2.08E-01
3b	Large Isolation Failures (liner breach)	1.18E-07	4.41E+06	5.19E-01
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A
7-CFE	Failures Induced by Phenomena (Early)	5.20E-07	5.08E+07	2.64E+01
7-CFL	Failures Induced by Phenomena (Late)	2.60E-05	1.63E+07	4.24E+02
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	2.60E-07	5.08E+07	1.32E+01
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	0.00E+00	5.08E+07	0.00E+00
CDF	**All CET End States (Including Intact Case)**	**5.20E-05**		**467.9**

TABLE 5.7-6A
POPULATION DOSE RISK AS A FUNCTION OF ACCIDENT CLASS (IP2 ALTERNATIVE EXTERNAL EVENTS EVALUATION CHARACTERISTIC OF CONDITIONS FOR 1 IN 15 YEAR ILRT FREQUENCY)

ACCIDENT CLASS (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (1/YR)	DOSE (PERSON-REM)	DOSE RISK (PERSON-REM/YR)
1	Containment Intact	2.14E-05	4.41E+04	9.44E-01
2	Large Isolation Failures (Failure to Close)	5.01E-08	6.51E+07	3.26E+00
3a	Small Isolation Failures (liner breach)	2.27E-06	4.41E+05	1.00E+00
3b	Large Isolation Failures (liner breach)	5.67E-07	4.41E+06	2.50E+00
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A
7-CFE	Failures Induced by Phenomena (Early)	5.01E-07	6.51E+07	3.26E+01
7-CFL	Failures Induced by Phenomena (Late)	2.51E-05	1.63E+07	4.08E+02
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	2.51E-07	6.51E+07	1.63E+01
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	0.00E+00	6.51E+07	0.00E+00
CDF	**All CET End States (Including Intact Case)**	**5.01E-05**		**464.9**

TABLE 5.7-6B
POPULATION DOSE RISK AS A FUNCTION OF ACCIDENT CLASS (IP3 ALTERNATIVE EXTERNAL EVENTS EVALUATION CHARACTERISTIC OF CONDITIONS FOR 1 IN 15 YEAR ILRT FREQUENCY)

ACCIDENT CLASS (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (1/YR)	DOSE (PERSON-REM)	DOSE RISK (PERSON-REM/YR)
1	Containment Intact	2.22E-05	4.41E+04	9.80E-01
2	Large Isolation Failures (Failure to Close)	5.20E-08	5.08E+07	2.64E+00
3a	Small Isolation Failures (liner breach)	2.35E-06	4.41E+05	1.04E+00
3b	Large Isolation Failures (liner breach)	5.88E-07	4.41E+06	2.59E+00
4	Small Isolation Failures (Failure to seal –Type B)	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A	N/A
7-CFE	Failures Induced by Phenomena (Early)	5.20E-07	5.08E+07	2.64E+01
7-CFL	Failures Induced by Phenomena (Late)	2.60E-05	1.63E+07	4.24E+02
8-SGTR	Containment Bypass (Steam Generator Tube Rupture)	2.60E-07	5.08E+07	1.32E+01
8-ISLOCA	Containment Bypass (Interfacing System LOCA)	0.00E+00	5.08E+07	0.00E+00
CDF	**All CET End States (Including Intact Case)**	**5.20E-05**		**470.7**

TABLE 5.7-7
COMPARISON TO ACCEPTANCE CRITERIA INCLUDING ALTERNATIVE EXTERNAL EVENTS EVALUATION CONTRIBUTION FOR IP2 AND IP3

Contributor	ΔLERF	ΔPerson-rem/yr	ΔCCFP
IP2 Internal Events	9.84E-8/yr	0.584/yr (0.62%)	0.84%
IP2 External Events	4.54E-7/yr	2.70/yr (0.58%)	0.91%
Indian Point 2 Total	**5.52E-7/yr**	**3.28/yr (0.59%)**	**0.89%**
IP3 Internal Events	1.26E-7/yr	0.751/yr (0.93%)	0.85%
IP3 External Events	4.71E-7/yr	2.80/yr (0.60%)	0.91%
Indian Point 3 Total	**5.96E-7/yr**	**3.55/yr (0.65%)**	**0.89%**
Acceptance Criteria	**<1.0E-6/yr**	**<1.0 person-rem/yr or <1.0%**	**<1.5%**

The 5.52E-07/yr increase in LERF for IP2 and the 5.97E-07/yr increase in LERF for IP3 due to the combined internal and external events from extending the ILRT frequency from 3-per-10 years to 1-per-15 years falls within Region II between 1.0E-7 to 1.0E-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 1.0E-5/yr.

From Table 4.2-1, the total LERF due to postulated internal event accidents is 1.16E-06/yr for IP2 and 1.25E-06/yr for IP3. From Table 5.7-5a for IP2, the base external events LERF can be derived from the Class 2, Class 3b, Class 7-CFE, and Class 8 contributions. From the individual contributions of 5.01E-08/yr + 1.13E-07/yr + 5.01E-07/yr + 2.51E-07/yr, this equates to 9.15E-07/yr. From Table 5.7-5b for IP3, the individual contributions of 5.20E-08/yr + 1.18E-07/yr + 5.20E-07/yr + 2.60E-07/yr result in a total base case LERF from external events of 9.50E-07/yr. The total LERF values for IP2 and IP3 using the alternative external events evaluation are then shown in Table 5.7-8.

TABLE 5.7-8
IMPACT OF 15-YR ILRT EXTENSION ON LERF FOR IP2 AND IP3

LERF CONTRIBUTOR	IP2 (1/YR)	IP3 (1/YR)
Internal Events LERF	1.16E-06	1.25E-06
External Events LERF	9.15E-07	9.50E-07
Internal Events LERF due to ILRT (at 15 years) (1)	1.23E-07	1.58E-07
External Events LERF increase due to ILRT extension (2)	4.54E-07	4.71E-07
Total	**2.65E-06/yr**	**2.83E-06/yr**

(1) Including age adjusted steel liner corrosion likelihood as reported in Table 5.7-2a for IP2 and Table 5.7-2b for IP3.

(2) As shown in Table 5.7-7. This did not include the age adjusted steel liner corrosion likelihood, but this was demonstrated to be a small contributor for IP2 and IP3.

As can be seen, the total LERF for IP2 is estimated as 2.65E-06/yr and for IP3 it is 2.83E-06/yr. These values are both less than the RG 1.174 requirement to demonstrate that the total LERF due to internal and external events is less than 1.0E-5/yr.

5.8 CONTAINMENT OVERPRESSURE IMPACTS ON CDF

For IP2 and IP3, ECCS NPSH calculations made in support of the GSI-191 effort [28, 29] confirmed that containment overpressure is not required to obtain adequate NPSH [30]. This is consistent with the PRA models which indicate there is no impact on CDF from the ILRT extension risk assessment.

- In IP-CALC-06-000231 [28], the NPSHA / NPSHR relationship for IP2 ECCS pumps was being evaluated. For conservatism in obtaining the NPSHA and NPSHR, the maximum volumetric flow rate was used. The greatest volumetric flow rate occurs when the least dense fluid is being pumped. This is at the highest temperature in the recirculation phase of the accident. For IP2, this temperature was 264.4 F which occurs at start of recirculation. Since 264.4 F is higher than 212 F, a boundary condition pressure of 37.6 psia is inputted. This is close to the saturation pressure at 264.4 F so there is essentially no containment overpressure being invoked. In other words, 264.4 F and 37.6 psia is basically equivalent to 212 F and 14.7 psia (0 psig).
- The same issue was addressed in IP-CALC-07-00054 [29] for the IP3 NPSHA / NPSHR evaluation. Again, to be most conservative with respect to NPSHA and NPSHR, the maximum volumetric flow rate has to be used. This entails that the highest temperature during recirculation applies. This is 242.8 F at commencement of recirculation. The saturation pressure at 242.8 F is close to 26.1 psia, which is the boundary condition pressure input in the calculation. Again, essentially no containment overpressure is being invoked since 242.8 F and 26.1 psia is basically equivalent to 212 F and 14.7 psia (0 psig).

6.0 SENSITIVITIES

6.1 SENSITIVITY TO CORROSION IMPACT ASSUMPTIONS

The results in Tables 5.2-2a(b), 5.3-1a(b), and 5.3-2a(b) 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. 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 cylinder, dome 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 6.1-1a for IP2 and in Table 6.1-1b for IP3. 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 3.68E-8/yr for IP2 and 4.72E-08/yr for IP3. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

TABLE 6.1-1A
STEEL LINER CORROSION SENSITIVITY CASES FOR IP2

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 FROM 3 IN 10 TO 1 IN 15 YEARS (PER YEAR)	
			TOTAL INCREASE	INCREASE DUE TO CORROSION
Base Case Doubles every 5 yrs	Base Case (1.0% Cylinder-Dome, 0.1% Basemat)	Base Case (10% Cylinder-Dome, 100% Basemat)	9.84E-08	1.16E-09
Doubles every 2 yrs	Base	Base	9.99E-08	2.63E-09
Doubles every 10 yrs	Base	Base	9.83E-08	9.68E-10
Base	Base	15% Cylinder-Dome	9.89E-08	1.62E-09

TABLE 6.1-1A
STEEL LINER CORROSION SENSITIVITY CASES FOR IP2

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 FROM 3 IN 10 TO 1 IN 15 YEARS (PER YEAR)	
			TOTAL INCREASE	INCREASE DUE TO CORROSION
Base	Base	5% Cylinder-Dome	9.79E-08	6.97E-10
Base	10% Cylinder-Dome, 1% Basemat	Base	1.09E-07	1.16E-08
Base	0.1% Cylinder-Dome, 0.01% Basemat	Base	9.74E-08	1.16E-10
LOWER BOUND				
Doubles every 10 yrs	0.1% Cylinder-Dome, 0.01% Basemat	5% Cylinder-Dome, 100% Basemat	9.73E-08	5.81E-11
UPPER BOUND				
Doubles every 2 yrs	10% Cylinder-Dome, 1% Basemat	15% Cylinder-Dome, 100% Basemat	1.34E-07	3.68E-08

TABLE 6.1-1B

STEEL LINER CORROSION SENSITIVITY CASES FOR IP3

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 FROM 3 IN 10 TO 1 IN 15 YEARS (PER YEAR)	
			TOTAL INCREASE	INCREASE DUE TO CORROSION
Base Case Doubles every 5 yrs	Base Case (1.0% Cylinder-Dome, 0.1% Basemat)	Base Case (10% Cylinder-Dome, 100% Basemat)	1.26E-07	1.49E-09
Doubles every 2 yrs	Base	Base	1.28E-07	3.37E-09
Doubles every 10 yrs	Base	Base	1.26E-07	1.24E-09
Base	Base	15% Cylinder-Dome	1.27E-07	2.08E-09
Base	Base	5% Cylinder-Dome	1.26E-07	8.95E-10
Base	10% Cylinder-Dome, 1% Basemat	Base	1.40E-07	1.49E-08
Base	0.1% Cylinder-Dome, 0.01% Basemat	Base	1.25E-07	1.49E-10
LOWER BOUND				
Doubles every 10 yrs	0.1% Cylinder-Dome, 0.01% Basemat	5% Cylinder-Dome, 100% Basemat	1.25E-07	7.47E-11
UPPER BOUND				
Doubles every 2 yrs	10% Cylinder-Dome, 1% Basemat	15% Cylinder-Dome, 100% Basemat	1.72E-07	4.72E-08

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 [3]. 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 failure mechanisms, the potential for undiscovered mechanisms, inaccessible 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 within containment. The base case methodology uses the Jeffrey's non-informative prior for the large leak size and the expert elicitation sensitivity study uses the results from 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 base case methodology (i.e., 10La for small and 100La for large) are used here. Table 6.2-1 illustrates the magnitudes and probabilities of a pre-existing leak in containment associated with the base case 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, including the input to expert elicitation as well as the results of the expert elicitation, are available in the various appendices of EPRI 1018243 [3].

TABLE 6.2-1
EPRI EXPERT ELICITATION RESULTS

LEAKAGE SIZE (LA)	BASE CASE MEAN PROBABILITY OF OCCURRENCE	EXPERT ELICITATION MEAN PROBABILITY OF OCCURRENCE [3]	PERCENT REDUCTION
10	9.2E-03	3.88E-03	58%
100	2.3E-03	2.47E-04	89%

The summary of results using the expert elicitation values for probability of containment leakage is provided in Table 6.2-2a for IP2 and in Table 6.2-2b for IP3. As mentioned previously, probability values are those associated with the magnitude of the leakage used in the base case evaluation (10La for small and 100La for large). The expert elicitation process produces a relationship between probability and leakage magnitude in which it is possible to assess higher leakage magnitudes that are more reflective of large early releases; however, these evaluations are not performed in this particular study.

The net effect is that the reduction in the multipliers shown above also leads to a dramatic reduction on the calculated increases in the LERF values. As shown in Table 6.2-2a for IP2, the increase in the overall value for LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is just 1.05E-08/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is just 4.40E-09/yr. As shown in Table 6.2-2b for IP3, the increase in the overall value for LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is just 1.34E-08/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is just 5.60E-09/yr. As such, if the expert elicitation probabilities of occurrence are used instead of the non-informative prior estimates, the change in LERF for IP2 and IP3 is within the range of a "very small" change in risk when compared to the current 1-in-10, or baseline 3-in-10 year requirement. Additionally, as shown in Table 6.2-2a for IP2 and Table 6.2-2b for IP3, the increase in dose rate and CCFP are similarly reduced to much smaller values. 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 values from the EPRI methodology [3], and yet are still conservative given the assumption that all of the Class 3b contribution is considered to be LERF.

TABLE 6.2-2A
IP2 ILRT CASES:
3 IN 10 (BASE CASE), 1 IN 10, AND 1 IN 15 YR INTERVALS
(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 (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	4.41E+04	7.82E-06	3.45E-01	7.71E-06	3.40E-01	7.64E-06	3.37E-01
2	6.51E+07	1.11E-08	7.23E-01	1.11E-08	7.23E-01	1.11E-08	7.23E-01
3a	4.41E+05	4.10E-08	1.81E-02	1.37E-07	6.03E-02	2.05E-07	9.05E-02
3b	4.41E+06	2.61E-09	1.15E-02	8.70E-09	3.84E-02	1.31E-08	5.76E-02
7-CFE	6.22E+07	7.37E-08	4.58E+00	7.37E-08	4.58E+00	7.37E-08	4.58E+00
7-CFL	6.87E+06	2.71E-06	1.86E+01	2.71E-06	1.86E+01	2.71E-06	1.86E+01
8-SGTR	6.51E+07	1.05E-06	6.80E+01	1.05E-06	6.80E+01	1.05E-06	6.80E+01
8-ISLOCA	6.51E+07	2.77E-08	1.80E+00	2.77E-08	1.80E+00	2.77E-08	1.80E+00
Total		1.17E-05	9.414E+01	1.17E-05	9.421E+01	1.17E-05	9.425E+01
ILRT Dose Rate from 3a and 3b		2.96E-02		9.86E-02		1.48E-01	
Delta Total Dose Rate(1)	From 3 yr	---		6.45E-02		1.11E-01	
	From 10 yr	---		---		4.62E-02	
3b Frequency (LERF)		2.61E-09		8.70E-09		1.31E-08	
Delta 3b LERF	From 3 yr	---		6.09E-09		1.05E-08	
	From 10 yr	---		---		4.40E-09	
CCFP %		33.00%		33.05%		33.09%	
Delta CCFP %	From 3 yr	---		0.05%		0.09%	
	From 10 yr	---		---		0.04%	

(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 due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

TABLE 6.2-2B
IP3 ILRT CASES:
3 IN 10 (BASE CASE), 1 IN 10, AND 1 IN 15 YR INTERVALS
(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 (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	4.41E+04	1.12E-05	4.96E-01	1.11E-05	4.90E-01	1.10E-05	4.86E-01
2	5.08E+07	3.99E-09	2.03E-01	3.99E-09	2.03E-01	3.99E-09	2.03E-01
3a	4.41E+05	5.27E-08	2.32E-02	1.76E-07	7.74E-02	2.64E-07	1.16E-01
3b	4.41E+06	3.36E-09	1.48E-02	1.12E-08	4.93E-02	1.68E-08	7.40E-02
7-CFE	3.17E+07	1.88E-07	5.97E+00	1.88E-07	5.97E+00	1.88E-07	5.97E+00
7-CFL	6.85E+06	2.17E-06	1.49E+01	2.17E-06	1.49E+01	2.17E-06	1.49E+01
8-SGTR	5.08E+07	9.77E-07	4.96E+01	9.77E-07	4.96E+01	9.77E-07	4.96E+01
8-ISLOCA	5.08E+07	1.93E-07	9.80E+00	1.93E-07	9.80E+00	1.93E-07	9.80E+00
Total		1.48E-05	8.099E+01	1.48E-05	8.108E+01	1.48E-05	8.114E+01
ILRT Dose Rate from 3a and 3b		3.81E-02		1.27E-01		1.90E-01	
Delta Total Dose Rate[1]	From 3 yr	---		8.29E-02		1.42E-01	
	From 10 yr	---		---		5.94E-02	
3b Frequency (LERF)		3.36E-09		1.12E-08		1.68E-08	
Delta 3b LERF	From 3 yr	---		7.84E-09		1.34E-08	
	From 10 yr	---		---		5.60E-09	
CCFP %		23.84%		23.89%		23.93%	
Delta CCFP %	From 3 yr	---		0.05%		0.09%	
	From 10 yr	---		---		0.04%	

(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 due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT 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 permanently 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 1.0E-06/yr and increases in LERF below 1.0E-07/yr. "Small" changes in risk are defined as increases in CDF below 1.0E-05/yr and increases in LERF below 1.0E-06/yr. Since the ILRT extension was demonstrated to have no impact on CDF for IP2 and IP3, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval for the base case with corrosion included for IP2 is 9.84E-08/yr (see Table 5.6-1a). In using the EPRI Expert Elicitation methodology, the change is estimated as 1.05E-08/yr (see Table 6.2-2a). Both of these values fall within the very small change region of the acceptance guidelines in Reg. Guide 1.174. For IP3, the increase is estimated at 1.26E-07/yr (see Table 5.6-1b), which is within the small change region of the acceptance guidelines in Reg. Guide 1.174. In using the EPRI Expert Elicitation methodology, the change is estimated as 1.34E-08/yr (see Table 6.2-2b), which is within the very small change region of the acceptance guidelines in Reg. Guide 1.174.
- The change in dose risk for changing the Type A test frequency from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all internal events accident sequences for IP2, is 0.584 person-rem/yr (0.62%) using the EPRI guidance with the base case corrosion case (Table 5.6-1a). The change in dose risk drops to 1.11E-01 person-rem/yr when using the EPRI Expert Elicitation methodology (Table 6.2-2a). For IP3, it is 0.751 person-rem/yr (0.93%) using the EPRI guidance with the base case corrosion case (Table 5.6-1b). The change in dose risk drops to 1.42E-01 person-rem/yr when using the EPRI Expert Elicitation methodology (Table 6.2-2b). The values calculated per the EPRI guidance are all lower than the acceptance criteria of ≤1.0 person-rem/yr or <1.0% person-rem/yr defined in Section 1.3.
- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen years including corrosion effects using the EPRI guidance (see Section 5.5) is 0.84% for IP2 and 0.85% for IP3. This value drops to less that 0.10% for IP2 and IP3 using the EPRI Expert Elicitation methodology (see Table 6.2-2a and Table 6.2-2b, respectively). This is below the acceptance criteria of less than 1.5% defined in Section 1.3.
- To determine the potential impact from external events, a bounding assessment from the risk associated with external events utilizing information from the IP2 and IP3 IPEEEs similar to the approach used in the License Renewal SAMA analysis was performed. As shown in Table 5.7-2a for IP2, the total increase in LERF due to internal events and the bounding external events assessment is 5.20E-07/yr. As shown in Table 5.7-2b for IP3, the total increase in LERF due to internal events and the bounding external events assessment is 5.70E-07/yr. Both of these values are in Region II of the Reg. Guide 1.174 acceptance guidelines.

- As shown in Table 5.7-4, the same bounding analysis indicates that the total LERF from both internal and external risks is 6.78E-06/yr for IP2 and 6.34E-06/yr for IP3, which are less than the Reg. Guide 1.174 limit of 1.0E-05/yr given that the ΔLERF is in Region II (small change in risk).
- Finally, since the external events assessment led to exceeding one of the two alternative acceptance criteria (i.e. greater than 1.0 person-rem/yr, an alternative detailed bounding external events assessment was also performed to demonstrate that the alternate 1.0% person-rem/yr criterion and the other acceptance criteria could still be met. In this case, as shown in Table 5.7-7 for IP2, the total change in LERF from both internal and external events was 5.52E-7/yr, the change in person-rem/yr was 3.28/yr representing 0.59% of the total, and the change in the CCFP was 0.89%. For IP3, the total change in LERF from both internal and external events was 5.97E-7/yr, the change in person-rem/yr was 3.55/yr representing 0.65% of the total, and the change in the CCFP was 0.89%. All of these calculated changes meet the acceptance criteria. As shown in Table 5.7-8, this assessment indicates that the total LERF from both internal and external risks is 2.65E-06/yr for IP2 and 2.83E-06/yr for IP3, which are less than the Reg. Guide 1.174 limit of 1.0E-05/yr given that the ΔLERF is in Region II (small change in risk).
- Including age-adjusted steel liner corrosion effects in the ILRT assessment was demonstrated to be a small contributor to the impact of extending the ILRT interval for IP2 and IP3.

Therefore, increasing the ILRT interval on a permanent basis to a one-in-fifteen year frequency is not considered to be significant since it represents only a small change in the IP2 and IP3 risk profiles.

<u>Previous Assessments</u>

The NRC in NUREG-1493 [6] has previously concluded the following:

- 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 IP2 and IP3 confirm these general findings on a plant specific basis considering the severe accidents evaluated, the containment failure modes, and the local population surrounding IP2 and IP3.

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Appendix A
PRA Technical Adequacy

Note that the information provided in this appendix was provided by Entergy personnel.

A.1 OVERVIEW

A technical Probabilistic Risk Assessment (PRA) analysis is presented in this report to help support an extension of the IP2 and IP3 containment Type A test integrated leak rate test (ILRT) interval to fifteen years.

The analysis follows the guidance provided in Regulatory Guide 1.200, Revision 2 [A.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. 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 report. The purpose of this appendix is to address the requirements identified in item 4 above. 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) are discussed in the following sections.

The risk assessment performed for the ILRT extension request is based on the current Level 1 and Level 2 PRA models of record. Information developed for the license renewal effort to support the Level 2 release categories is also used in this analysis supplemented by additional calculations to more appropriately represent the intact containment case in the ILRT extension risk assessment.

Note that for this application, the accepted methodology involves a bounding approach to estimate the change in the LERF from extending the ILRT interval. Rather than exercising the PRA model itself, it involves the establishment of separate evaluations that are linearly related to the plant CDF contribution. Consequently, a reasonable representation of the plant CDF that does not result in a 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.

As further discussed below, the PRA models used for this application are the latest models, which were released in November 2011 (for IP2) and November 2012 (for IP3). There are no significant plant changes (design or operational practices) that have not yet been incorporated in those PRA models.

A discussion of the Entergy model update process, the peer reviews performed on the IP2 and IP3 models, the results of those peer reviews and the potential impact of peer review findings on the ILRT extension risk assessment are provided in Section A.2. Section A.3 provides an assessment of key assumptions and approximations used in this assessment and Section A.4 briefly summarizes the results of the PRA technical adequacy assessment with respect to this application.

A.2 PRA UPDATE PROCESS AND PEER REVIEW RESULTS

A.2.1 Introduction

The Indian Point Unit 2 (IP2) and Unit 3 (IP3) Probabilistic Risk Assessment (PRA) models used for this application [A.2 and A.3] are the most recent evaluations of the IP2 and IP3 risk profiles for internal event challenges. The IP2 and IP3 PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause failure events. The PRA model quantification process is based on the event tree and fault tree methodology, which is a well-known methodology in the industry.

Entergy employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Entergy nuclear power plants. 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 IP2 and IP3 PRA models.

A.2.2 PRA Maintenance and Update

The Entergy risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plant. This process is defined in the Entergy fleet procedure EN-DC-151, "PSA Maintenance and Update" [A.4]. This procedure delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Entergy nuclear power plants. In addition, the procedure also defines the process for implementing regularly scheduled and interim PRA model updates, and for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.). To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model. Potential PRA model changes resulting from these reviews are entered into the Model Change Request (MCR) database, and a determination is made regarding the significance of the change with respect to current PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years, and
- Industry standards, experience, and technologies are periodically reviewed to ensure that any changes are appropriately incorporated into the models.

In addition, following each periodic PRA model update, Entergy performs a self-assessment to assure that the PRA quality and expectations for all current applications are met. The Entergy PRA maintenance and update procedure requires updating of all risk informed applications that may have been impacted by the update.

A.2.3 Regulatory Guide 1.200 PWROG Peer Review of the IP2 and IP3 Internal Events PRA Models

Both the IP2 and IP3 internal events models went through a Regulatory Guide 1.200 PWR Owners Group peer review using the NEI 05-04 process.

The IP2 PRA internal events model peer review was performed in December 2009, and used the American Society of Mechanical Engineers PRA Standard RA-Sb-2005, and Regulatory Guide 1.200 Revision 1. The IP3 PRA internal events model peer review was performed in December 2010. Since the IP3 peer review was later, it used RA-Sa-2009 (the American Society of Mechanical Engineers / American Nuclear Society Combined PRA Standard) and Regulatory Guide 1.200 Revision 2. As noted in the forward to the combined standard, the primary purpose, in addition to combining internal and external events into a single standard, was to ensure consistency in format, organization, language, and level of detail. It was also noted that, among the criteria observed in assembling the component Standards were:

(a) the requirements in the Standards would not be revised or modified

(b) no new requirements would be included

An internal comparison of the ASME standard to the combined ASME / ANS standard confirmed that there were few substantive changes to the internal events portion of the standard, although the expected level of documentation was increased in some cases.

The IP2 and IP3 PRA peer reviews addressed all the technical elements of the internal events, at-power PRA:

- Initiating Events Analysis (IE)
- Accident Sequence Analysis (AS)
- Success Criteria (SC)
- Systems Analysis (SY)
- Human Reliability Analysis (HR)
- Data Analysis (DA)
- Internal Flooding (IF)
- Quantification (QU)
- LERF Analysis (LE)
- Maintenance and Update Process (MU)

During the IP2 and IP3 PRA model peer reviews, the technical elements identified above were assessed with respect to Capability Category II criteria to better focus the Supporting Requirement assessments.

A.2.4 Peer Review Results

The ASME PRA standards used for the IP2 and IP3 peer reviews each contained a total of 326 numbered supporting requirements. A number of the supporting requirements were determined to be not applicable to the IP2 or IP3 PRA (e.g., BWR related, multi-site related). Of the applicable supporting requirements, 95% were satisfied at Capability Category II or greater for IP2, and 97% were satisfied at Capability Category II criteria or greater for IP3.

The Facts and Observations (F&Os) for the IP2 PRA peer review are provided in the report, entitled, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Indian Point 2 Nuclear Power Plant Probabilistic Risk Assessment" [A.5]. Of the 41 Facts and Observations (F&Os) generated by the Peer Review Team, 21 were considered Findings.

The Facts and Observations (F&Os) for the IP3 PRA peer review are provided in the report, entitled, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Indian Point 3 Probabilistic Risk Assessment" [A.6]. Of the 68 Facts and Observations (F&Os) generated by the Peer Review Team, 11 were considered Findings.

As a result of the Regulatory Guide 1.200 PWROG peer reviews, all the F&Os (other than best practices) were identified as potential improvements to the IP2 and IP3 PRA models or documentation and were entered into the Entergy Model Change Request (MCR) database. Tables A.2-1 and A.2-2 contain the findings resulting from the peer review of each unit, the status of the resolution for each finding and the potential impact of each finding on this application. In summary, a majority of the findings were related to documentation and have no material impact. As shown, almost all findings have been resolved and incorporated into the updated model and/or documentation. Resolution of the few open peer review findings is expected to have, at most, a minor impact on the model and its quantitative results and no significant impact on the conclusions of this application.

In resolving the IP3 peer review findings, several additional internal flooding sources were identified as not being addressed in the original internal flooding analysis report. Most of those sources involved fire protection piping, but they also included auxiliary component cooling water (ACCW) piping in the fan house and short sections of component cooling water (CCW) piping in a pipe chase in the foyer outside the charging pump rooms. These additional sources were included in the final model used for this application.

A.2.5 External Events

Although EPRI report 1018243 [A.7] recommends a quantitative assessment of the contribution of external events (for example, fire and seismic) where a model of sufficient quality exists, it also recognizes that the external events assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval. Since the most current external events models for IP2 and IP3 are those embodied in the IPEEE, a multiplier was applied to the internal events results based on the IPEEE, similar to that used in the SAMA analysis [A.8 and A.9]. This is further discussed in Section 5.7 of the risk assessment.

A.2.6 Summary

The IP2 and IP3 PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that these PRA models are suitable for use in the risk-informed process used for this application.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-3	Appendix A1, Section 3.4, "Other Initiating Events" states 'Other plant-specific initiators and event precursors were also investigated using an FMEA of plant systems as discussed below and this was reviewed with plant personnel to verify expected plant response.' It is not clear that interviews were conducted.	IE-A8	Appendix A1, Section 3.4, "Other Initiating Events" states 'Other plant-specific initiators and event precursors were also investigated using an FMEA of plant systems as discussed below and this was reviewed with plant personnel to verify expected plant response.' It is not clear that interviews were conducted.	Document the interviews	**OPEN** This is a documentation issue. Although discussions were held with plant personnel, no formal interview form or format was used. This remains open as a documentation improvement item for the next update.	**No Impact** This is a documentation enhancement issue.
1-7	Not met since the frequencies were not weighted by the fraction of time the plant was at power.	IE-C5	The SR requires that the IE frequencies be weighted by the plant availability. This has not been done for IP2 initiating events.	Weight the initiating event frequency time by the fraction of time the plant was at power.	**OPEN** While we agree that the wording in the SR itself indicates that weighting should be done, the ASME standard acknowledges that the SR wording is somewhat unclear and provides a detailed note of explanation (Note 1 of the SR). Entergy believes that using the annual average model, which Note 1 acknowledges should not include the weighting factors, is the appropriate baseline model in the absence of an all modes model. We do agree, as the standard states, that an all modes model should account for the time in each operating state. Entergy does not have an all modes model at this time. We believe that tying risk values to plant availability without an all modes model can potentially provide inappropriate risk insights to non-PSA personnel. It does not apply any risk to other operating states. Therefore, we believe that at the least, our current model meets the SR, when taken in concert with the associated Note 1.	**No significant impact** The current approach provides a slightly conservative result, and use of the stipulated weighting approach would have no significant impact on this application.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-8	While the documentation of the Success Criteria is detailed with sufficient information to support the model development, the lack of references to supporting documents for a variety of assumptions and sections makes the review difficult and the ability to maintain the model based upon plant changes and analysis revisions very difficult to track and change. Examples are: 1) RCS peak pressure within 120 seconds of an ATWS 2) The normal relief flow through each PORV valve is 179,000 lb/hr; the maximum flow is 210,000 lb/hr Note that these are simply a couple of examples of a more prevalent issue.	SC-C1	The current documentation poses a potential problem in facilitating PRA applications, upgrades, and peer review due to the significant amount of information included that is not traceable.	Provide basis for parameters, limits, setpoints, etc.	**Resolved** Additional references/basis for parameters, limits, and setpoints were added to Section B1.3.2, "Level 1 Assumptions" and other pertinent sections of the success criteria analysis notebook.	**No Impact** Documentation issue - incorporated in final project file for the model used for this application.
1-11	Attachment E summarizes the calculation of initiating event frequencies but there must be a table that shows the actual calculations using generic, plant-specific, and Bayesian updating. It would be helpful to include this table.	IE-C4 IE-C5	Attachment E summarizes the calculation of initiating event frequencies but there must be a table that shows the actual calculations using generic, plant-specific, and Bayesian updating. It would be helpful to include this table.	Produce a table which shows the actual calculations using generic, plant-specific, and Bayesian updating	**Resolved** Added a table showing a sample calculation to enhance Appendix A1 of the update report. The calculations used to develop the IE frequencies are contained in the EXCEL files that are part of the IP2 model update project files and are retained for future reviews, updates or applications. This issue is only a matter of the extent and the details of the calculations extracted and made part of the written report. Also note that the methodology used for these calculations was discussed in Appendix A1, Section 11 and the results were summarized in Attachment E.	**No Impact** Documentation issue - incorporated in final project file for the model used for this application.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-13	No definition or criteria for the definitions of failure modes, and success criteria were identified in the review of the Data analysis package.	DA-A2	The criteria to establish the definitions of SSC boundaries, failure modes, and success criteria in a manner consistent with corresponding basic event definitions in Systems Analysis are required per the SR. In this case SSC boundaries were discussed and examples provided. However, there was no similar documentation for the failure modes and success criteria	Provide documentation regarding the failure modes to consider for evaluation of the data analysis and the associated success criteria. (It is noted that Attachment 2 of Appendix D0, identifies many of the issues for consideration in relation to this SR.)	**Resolved** This is a documentation issue. The current process satisfies the requirements of this SR. The boundaries, failure modes, and success criteria considered in the Data Analysis are consistent with those used for each system to match the failure modes, common cause and boundaries of unavailability events. The data analysis notebook discusses this (for example, see Appendix D1, sections 1.4 and 3.1 thru 3.3 and 4.1, 4.3 and 4.6) and shows that these are all addressed in the updated plant model. App. D1, Attachment A includes discussions and definitions of component boundaries related to component failure modes and how this was considered in the data analysis. This is consistent with Appendix E, Table E0.1-3 which lists the failure modes and associated codes that are used in the model. All modeled basic events are captured in the fault trees and the associated model data base with codes corresponding to this table and the Data Analysis is shown to match the failure modes and boundaries of these events. In the associated System Notebook, each fault tree is discussed and the overall system success criteria in the model are summarized.	**No Impact** Documentation issue - incorporated in final project file for the model used for this application.
1-14	Accident sequences that reach and remain in a stable state for 24 hours are assumed to be successfully mitigated. This can be interpreted to mean that the mission time is 24 hours after reaching a stable state. This statement should indicate that the accident sequence is considered mitigated when a stable state without core damage is reached.	AS-A8	DEFINE the end state of the accident progression as occurring when either a core damage state or a steady state condition has been reached	Rewrite the statement to indicate that the accident sequence is mitigated when a stable state without core damage has been reached. The mission time for this is usually 24 hours.	**Resolved** The statement referred to in the finding, which exists in Section 4 of the main report and in Appendix F1.0, has been revised to read: "Accident sequences that reach a stable state within 24 hours and remain in that state for the 24 hour mission time after the initiating event are assumed to be successfully mitigated. It is assumed that sufficient additional resources exist and sufficient time is available by that time to respond to any additional challenges."	**No Impact** Documentation issue - incorporated in final project file for the model used for this application.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-16	SR is MET, however, three system packages in which the section relating to spatial dependencies had no conclusion as to whether a spatial dependency exists (e.g. Control Building HVAC, Primary Water, AFWP Building Ventilation)	SY-B8	Walkdowns were documented as required for this SR. However, this is a documentation issue.	Provide conclusion of walkdown in all systems packages.	**Resolved** The walkdown records for the systems noted in the finding (Control Building HVAC, Primary Water and AFW Building Ventilation systems) have been reviewed and no spatial dependencies have been identified. The conclusion has been added to each of those system notebooks under Section 1.5 "LOCATION AND SPATIAL DEPENDENCIES". The remaining system notebooks already contain this conclusion.	**No Impact** Documentation issue - incorporated in final project file for the model used for this application.
1-18	Not Met CC II/III due to the lack of discussion and documentation relating to examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate	DA-D4	A review of the Update Spreadsheet in support of the Bayesian analysis reflects a single failure in which the posterior mean fell outside the uncertainty bound of the prior distribution.	Evaluate the posterior data in relation to the uncertainty bounds of the posterior and prior uncertainties to address discrepancies and document the issue such that the discrepancies (if they exist) can be explained or resolved.	**Resolved** Revised App. D1 and Data Analysis spreadsheet to follow the same approach used for IP3 and clarify that the requirement in SR DA-D4 to "check that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data" was performed. The discrepancies between the generic and the updated means were identified and evaluated and all were found to be reasonable based on the nature of the Bayesian update algorithm, the number of failures and the available plant data. Appendix D1, Section 3.6 was revised to discuss the approach. These statistical tests satisfy the requirements of DA-D4.	**No Impact** No change was required to the posterior data set.
1-19	There is no evidence that miscalibration of equipment that provided initiation signals for standby pumps were analyzed. Section H1.0 states: 'This review did not identify any Human Failure Events (HFEs) that are not already accounted for as possible failure modes in the Human Reliability analysis (HRA).'	HR-C2	INCLUDE those modes of unavailability that, following completion of each unscreened activity, result from failure to restore (b) initiation signal or set point for equipment start-up or realignment	Analyze miscalibration of equipment that provided initiation signals for standby pumps.	**Resolved** Comment incorporated. Additional pre-initiator human failure events (HFEs) were added to the model to represent miscalibration errors. See SAS system notebook, Table 1.2 Pre-Initiator Human Failure Events (HFEs) Screening.	**No Impact** Change incorporated in model used for this application.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-20	A review of the CCF in the System Work Packages (i.e. AFW) reveals that the Common Cause names listed do not match the common cause names in the model and data analysis package. (Example: FW406, FW-CCFS-AFWPM, etc.)	SY-B4	Naming convention should match in all references. This issue does not affect results since the model names and the data analysis names are consistent.	Correct the naming convention in the System Packages to match the model.	**Resolved** The common cause basic event names in the AFW System Work Packages have been corrected and now match the basic event names used in the AFW system fault tree model and data analysis.	**No Impact** Documentation issue - incorporated in final project file for the model used for this application.
1-23	In the Scope of Analysis it is stated: 'In this analysis, all causes of flooding were considered except plant-specific maintenance activities—the contribution of normal maintenance to flooding is included in the rupture frequency data used.' The flood frequencies in the EPRI flood guideline do not include maintenance.	IFSO-A4	For each potential source of flooding, IDENTIFY the flooding mechanisms that would result in a release. INCLUDE: (a) Failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. (b) Human-induced mechanisms that could lead to overfilling tanks, diversion of flow-through openings\ created to perform maintenance; inadvertent actuation of fire-suppression system (c) Other events resulting in a release into the flood area	Include maintenance induced flooding in the flood initiator frequencies	**Resolved** A search of the IP2 condition reporting system was performed for a period of 15 years for the Internal Flooding Analysis. No significant internal flooding events (including maintenance induced), were identified which would significantly alter the generic data.	**No Impact** No changes to the flooding frequency values were required.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-24	IDENTIFY the characteristic of release and the capacity of the source. INCLUDE the pressure and temperature of the source.	IFSO-A5 IFSO-A6	There is no documentation that identifies the pressure and temperature of the source.	Identify the pressure and temperature of the source.	**Open** This is a documentation issue. While Appendix C does not specifically identify the pressure and temperature of the sources, the analysis did document that the maximum flow rate resulting from a guillotine rupture was determined as well as lesser calculated release rates. A range of release sizes consistent with the available EPRI pipe rupture frequency data were, in fact, considered and a flow rate and frequency of occurrence derived for each. By this means, the size and frequency of possible releases were matched as required for the quantitative determination of the consequences of internal flooding. This remains an open finding, pending enhancement of the documentation regarding the pressures and temperatures of the ruptured systems to meet the letter of the SR.	**No Impact** This is a documentation issue. The description in Appendix C will be enhanced during the next update.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-26	Capability categories met. Latest versions of recognized generic data sources were used. Generic data for unavailability were not used. Note: The analysts ensured, to the extent possible, that the parameter definitions and component boundaries were consistent between the model and the data source. Appendix D notes that mismatches may be present, but that any such instances would be conservative because the generic data would include subcomponents that are treated separately in the model. Note: The opening paragraph in Attachment 1 indicates: 'The boundary definitions used in the model may need to be modified depending on the generic database and should be clearly defined so that the failure modes in the model match those in the generic databases.' Apparently, this was not done in all cases - as noted above.	DA-C1	It would be helpful to indicate instances in which the generic data and the model do not match. As currently documented, it is not clear how often this occurs or how significant mismatches of this type might be. Note: the EDG load output breakers are identified specifically in the text as being one area of mismatch. If this is the only instance, then this should be clarified.	Provide documentation regarding the failure modes to consider for evaluation of the data analysis and the associated success criteria. (It is noted that Attachment 2 of Appendix D0, identifies many of the issues for consideration in relation to this SR.)	**Resolved** Appendix D1 was revised to clarify that any mismatches are due to discrepancies in the generic data sources. Added the following wording to section 1.4 to address boundaries and other issues; "Consistent with System Analysis requirements, the failure rates, common cause failure events and unavailability events were identified from the system fault trees to be consistent with corresponding systems analysis definitions, success criteria and boundaries (to the extent practical considering the differences in the boundary definitions in the generic and common cause databases). Component failure data was matched to corresponding events in system fault trees. Failure modes that are in the system models were mapped to corresponding basic event Type Codes and other events used in CAFTA (common cause failure and maintenance unavailability events)." Also revised Attachment A, section 1.0, item 2 to add; "Note that the boundaries provided below are consistent with those used in NUREG/CR-6928, however they are not defined in the same manner or to the same level of detail as they are in the NRC CCF database which may result in overlaps in the boundaries that could lead to conservative estimates for the CCF failures". No additional documentation or evaluation of the data analysis is required to satisfy this requirement.	**No significant impact** At most, this may result in a slight conservatism as noted in the disposition.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-27	Met for CCI but not CCII; Section 3.7 System Unavailability Due to Testing and Maintenance discusses that 5 years of unavailability data was collected via the Maintenance Rule program. If no Maintenance Rule or plant records were available for a particular component, generic data from NUREG/CR-6928 were used to estimate unavailability.	DA-C13	Appendix D1, section 3.7 says 'If no Maintenance Rule or plant records were available for a particular component, generic data from NUREG/CR-6928 were used to estimate unavailability.'	Document the interviews used to meet this requirement	**Resolved** As demonstrated in the EXCEL file used for the data update, the population of components for which Maintenance Rule (MR) unavailability data did not exist was limited to the Appendix R Diesel Generator and a few MR non-risk significant systems. The Appendix R diesel has only been in service a limited time and the System Engineer confirmed that there were no unavailable hours that could be applied for the update. The Maintenance Rule Coordinator and/or the appropriate System Engineers were queried regarding the other systems for which MR availability was not monitored but were unable to provide reliable estimates due to the lack of monitoring data. As a result, generic data was applied to these system components. Since the discussions with plant personnel did not yield useful information and could not be used "to generate estimates" for unavailability, additional documentation of those discussions would be of little additional value and was not generated.	**No Impact** This was a documentation issue since there were no additional insights available from plant personnel.
2-2	Capability Category I met. Documentation in Appendix D1 was not sufficient to determine if it was necessary to decompose surveillance test data into sub-elements and whether this was done.	DA-C10	Discussion in Appendix D was not explicit enough to know whether Cat II was met.	Add discussion to further explain whether this SR was met at Cat II.	**Resolved** Appendix D1, Section 3.4 was enhanced to clarify that failure modes were not decomposed into sub-elements. Therefore, Appendix D does not address decomposition of failure modes and it was not necessary to perform additional reviews of surveillance tests to address sub-element specific data.	**No Impact** Documentation issue - incorporated in final project file for the model used for this application.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
3-2	Each System Notebook contains Table B-2a Supporting Requirements for HLR-SY-A that states under SY-A20 something such as this for CCW: 'The Component Cooling Water System by its design function removes heat from containment. Therefore, the Component Cooling Water System is fully capable of providing heat removal. Therefore, no further analysis is required to support this function.' However it is not clear that analyses were done to take credit for equipment associated with recirc inside containment.	SY-B14 SY-A22	TAKE CREDIT for system or component operability only if an analysis exists to demonstrate that rated or design capabilities are not exceeded.	Provide analysis that the equipment can function beyond design basis environment.	**Resolved** The model only takes credit for component operability based on design or rated capability and does not assume or take credit for operation beyond design basis capability unless specific calculations and evaluations were available, as noted in the system notebooks for AFV, CB HVAC, EDGV. Clarification was provided in the system notebooks, as required, to revised wording of "Harsh Environments" under section 1.5 and in Table B-2a for how SY-A20 is met (see the other various system notebooks including CCW, CVCS, HHSI, LHSI, IAS, EDG, SWS).	**No Impact** Documentation issue - incorporated in final project file for the model used for this application.
3-4	There is no problem with the generic data or the Bayesian updating process used. The issue is the calculation of 'realistic parameter estimates' using plant specific data since only EPIX / Maintenance Rule information was used.	DA-D1	Issue centers on the calculation of 'realistic parameter estimates' using plant specific data since only EPIX / Maintenance Rule information was used.	Calculate realistic parameter estimates using plant specific data.	**Resolved** Revised failure identification to include plant failures not included in EPIX data as explained in revised Appendix D1, Section 3.5. Entergy fleet procedures and fleet standards address EPIX reporting and confirm that all Maintenance Rule (MR) functional failures require an EPIX report. They also require all failures of high critical components to be included in EPIX reporting, which includes failures that may cause a trip or impact plant operation, even of non-risk significant operating systems within MR scope that might be monitored under plant criteria and might not otherwise be captured. These requirements ensure that failures of all modeled components are captured in the EPIX data used for the PSA model. The only exceptions are failures of high critical components that occurred prior to 2007, when these procedures were implemented. Those failures were obtained from specific plant records and included in the update. No further action is required to satisfy this requirement.	**No Impact** No changes to the data analysis were required.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
4-1	Met CC II/III Based upon a thorough analysis of the generic data using plant specific data for Bayesian updating. However, there is a lack of discussion and documentation relating to examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm these inconsistencies are appropriate	DA-D4	A review of the Update Spreadsheet in support of the Bayesian analysis reflects a single failure in which the posterior mean fell outside the uncertainty bound of the prior distribution.	Evaluate the posterior data in relation to the uncertainty bounds of the posterior and prior uncertainties to address discrepancies and document the issue such that the discrepancies can be explained or resolved.	**Resolved** The associated data analysis spreadsheet was revised to allow the discrepancies between the generic and the updated means to be identified and evaluated and all were found to be reasonable based on the nature of the Bayesian update algorithm, the number of failures and the available plant data. Appendix D1, Section 3.6 revised to clarify that the requirement in SR DA-D4 to "check that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data" was performed. These statistical tests satisfy the requirements of DA-D4.	**No Impact** No change was required to the posterior data set.
4-2	This SR is Not MET. The use of EPIX as the basis for plant related failures associated with PRA modeled components is insufficient to ensure that all failures are captured. EPIX captures Maintenance Rule Functional Failures and Critical component failures (post 2007). Therefore, this database is limited in scope. Also it should be considered that the Maintenance Rule will not capture all failures associated with non-risk significant systems. Therefore, this data is also not inclusive.	DA-D1 DA-D4	It is not apparent that all plant specific failures associated with PRA related components have been captured in the data review for this model update.	Perform a more extensive review of the plant specific failures to ensure that the data is complete. (Note: should it be determined that the Indian Point EPIX database does actually include all PRA modeled component failures, this F&O can be dispositioned as such).	**Resolved** See disposition for finding 3-4.	**No Impact** No changes to the data analysis were required.
4-3	Documentation of the data analysis is not complete due to the lack of any reference to the basis for the data results. It was noted during the review that the data analysis is actually calculated using spreadsheets; however, those spreadsheets are not part of the data analysis package.	DA-E1	Supporting files were provided during the review that contained critical information relating to the data analysis. This information in the form of an Excel Spreadsheet is not included in the Data Analysis package and is not referenced by the package.	Incorporate the spreadsheet into the document or as a reference in order to ensure traceability of the analysis and inputs for the analysis. Also include guidance on the use of the information contained in within the spreadsheet.	**Resolved** Revised Appendix D1, Section 3.6 to include reference to the applicable spreadsheets along with discussion of how they are the basis for the results. The spreadsheets are also retained in the project files that are maintained available for PRA applications, upgrades, and future reviews. An example of the calculations in the Excel files was added to Appendix D. No further action is required to satisfy the requirements of this SR.	**No Impact** Documentation issue - incorporated in final project file for the model used for this application.

TABLE A.2-1

SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP2 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
4-4	The model uses a single value for RPS in relation to the ATWS tree and certain initiating Events. This RPS module for the ATWS logic is quantified using the RPS fault tree. Although modularization of initiating events allows for the determination of risk significance of the initiator, the use of this module restricts the usability of the model for risk significance determination for those components associated with RPS.	QU-B10	The modularization of RPS in the ATWS logic precludes the ability for risk significant determinations of support systems and components within RPS.	Incorporate the RPS fault tree system into the ATWS logic in a manner that allows results interpretation of individual events.	**Resolved** The RPS is a somewhat unique system, and while we agree that the modeling of RPS is not fully consistent with this SR, we disagree that this finding warrants the SR not being met. In particular, the RPS is a fail-safe system. As such, loss of a support system does not materially impact the reliability of the RPS. Although the shunt trip function does rely on 125V dc power, the increase in unreliability of the RPS associated with unavailability of dc power is negligible. In addition, regarding the modeling of transmitters and trip relays, it should be noted that the RPS fault tree, which is consistent with NUREG/CR-5500 (Volume 2), is conservative in that it only credits two trip signals (overpower delta T and pressurizer high pressure). Individual tests impacting the RPS are addressed for online maintenance by adjusting the top event for RPS unreliability accordingly. Furthermore, the limited applicability of the finding should not preclude the SR from being met.	**No Impact** Use of a single value for RPS unavailability has no impact on this application.

TABLE A.2-2
SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP3 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-11	Appendix C1 of IP-RPT-10-00023, Rev. 0 provides a high to medium level summary of the flood scenarios, and provides greater depth in some areas. Analysis details available to the peer review team such as flooding calculations, were not sufficient to support upgrades and would have to be obtained or reproduced for future model changes. The documentation also lacks in reference to quantification input documentation (initiator specific flag files)	IFSN-B1 IFSN-B2	Analysis details available to the peer review team such as flooding calculations, were not sufficient to support upgrades and would have to be obtained or reproduced for future model changes. The documentation also lacks in reference to quantification input documentation (initiator specific flag files)	Provide required documentation	**Resolved** The backup spreadsheets have been obtained as well as the software used for flood level calculations, instructions for use of this software and material that supports its application. This additional documentation was included in the final model documentation package. Initiator specific flag files are contained in the electronic files included in the model update documentation package. A list of flag files was also added to the internal flooding notebook.	**No impact** The backup required to support future model updates and applications is now in the project documentation.
1-12	The walkdown notes in Appendix A of IP-RPT-10-00023, Rev. 0, Appendix C.1 note the general location of each SSC with respect to its room and elevation as well as its submergence height. Some additional general locational information is sometimes identified in Section 4.2 of IP-RPT-10-00023, Rev. 0, Appendix C.1. For example, it may state that a flood source may impact one but not both trains of equipment; specifics are not given as to why both cannot be impacted (e.g., shielding, curbs, etc.), but the information implies the impact of spatial information. There is no specific physical location information related to spray type failures found in the documentation. SSCs are only identified locationally by their flood area and elevation. It cannot be determined which SSCs in any area are susceptible to spray from any specific spray source.	IFSN-A5	There is no specific physical location information found in the documentation for SSCs other than flood area and elevation. Therefore, it cannot be determined which SSCs in any area are susceptible to spray from any specific spray source. In the scenario development it identifies which equipment is impacted by spray, but it cannot be determined how that information was obtained or if it is correct.	For SSCs susceptible to spray failure (also see F&O 2-3), ensure sufficient relational location information between the target SSC and spray sources are provided so that a determination can be made as to whether the SSCs can be damaged by each potential spray source.	**Resolved** Additional discussion was added to the walkdown Appendix to support the spray impacts included in the model. This includes reference to environmental qualification documents where these were used as a basis for stating that equipment would not be vulnerable to spray damage. A conservative separation criterion of 30 feet was used in examining the potential for spray impacts in the analysis. The composite piping and general arrangement drawings were scrutinized to ascertain whether equipment could be sprayed should a line or other piece of equipment rupture. The text of the report has been changed to note this. Providing additional specific location information within the model documentation will be considered to support future updates but is considered a documentation enhancement issue with no expected impact on the analysis.	**No impact** Additional information has been included in the updated model documentation.

TABLE A.2-2
SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP3 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
1-15	The initiating event frequencies are not weighted by the fraction of time the plant is at power. Section 10.9 of Appendix A0 provides guidance to account for plant availability in initiating event calculations. Section 4.0 of Appendix A1 states that the availability factor for the data update period was calculated. However, the calculated value is not incorporated into the initiating event or final CDF results.	IE-C5	The initiating event frequencies are not weighted by the fraction of time the plant is at power.	Include the plant availability factor in the calculation of initiating event frequencies.	**Open** While we agree that the wording in the SR itself indicates that weighting should be done, the ASME standard acknowledges that the SR wording is somewhat unclear and provides a detailed note of explanation (Note 1 of the SR). Entergy believes that using the annual average model, which Note 1 acknowledges should not include the weighting factors, is the appropriate baseline model in the absence of an all modes model. We do agree, as the standard states, that an all modes model should account for the time in each operating state. Entergy does not have an all modes model at this time. We believe that tying risk values to plant availability without an all modes model can potentially provide inappropriate risk insights to non-PSA personnel. It does not apply any risk to other operating states. Therefore, we believe that at the least, our current model meets the SR, when taken in concert with the associated Note 1.	**No significant impact** The current approach provides, at most, a slightly conservative result in comparison to use of the stipulated weighting approach and would have no significant impact on this application.
3-7	The effects of the flood on PSFs were not specifically addressed in the HRA analysis.	IFQU-A6	Limited flooding-related human actions are included in the HRA discussion in Appendix H, but there is no mention of any effects of the flood on PSFs.	Discuss flood effects on PSFs and make adjustments to the HRA analysis if needed.	**Resolved** No short term isolation actions were credited in the flooding analysis. The only significant field action credited in the internal events model that could be impacted by the plant conditions associated with flooding was alignment of alternate cooling to the charging pumps on loss of CCW for certain specific CCW failure locations. The model has been updated to address that concern, and assumes that operator action is precluded by a break in the location that would impact that action.	**No impact** As discussed in the disposition, the only potential for a flooding impact on the modeled operator actions has been addressed in the updated model used for this application.

TABLE A.2-2
SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP3 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
4-14	Failure modes and success criteria defined in Systems Analysis are consistent with the Data Analysis. This SR also asks for establishing consistent SSC boundaries between the system level analysis and the data analysis. Reviewed Appendix E6 and E27 of the systems notebooks and Appendix D for the Data Analysis. Below is a list of issues identified: 1. System notebooks do not define the component boundaries. The component boundaries are defined by the generic failure rate data source with limited discussions on plant-specific SSC features and modeling considerations. 2. The guidance document Appendix D0 Section 5.10 states 'Assure the component boundaries established in the generic data match those defined in the PSA model. Make adjustments or justify differences'. Also, Attachment 4, Section 3.0 of the same document states that CCF boundaries are dictated by the fault tree modeling. However, the component boundaries defined for failure rate and CCF data do not match. The justification for using the data that way is that it is the conservative to do so. It is true that this approach is conservative for Emergency Diesel Generators, but it may not be conservative for other cases like batteries and battery chargers where CCF of output breakers are not modeled.	DA-A2 DA-D6	Based on the documents reviewed and the issues identified, component boundaries are not consistent among failure rate, CCF and unavailability data. Plant-specific features need to be considered for boundary definitions. It is possible to ensure that the inconsistent boundary definitions result in conservative results, but realistic rather than conservative results is ideal. CCF events tend to dominate system level cutsets and conservative CCF basic event values may mask other important components in a system.	As described in Sections 5.10 and 6.3.11 of Appendix D0, assure component boundaries defined in failure rate and CCF data match the PSA model. Assure the boundaries used in the test and maintenance data is consistent with the PSA model. Make adjustments or provide justification for any mismatch identified. Review plant-specific CCF experience for consistency to meet SY DA-D6 requirements.	**Resolved** This was a documentation issue. The level of modeling in the IP3 update required use of various databases since not all databases provided data for the components included in the model. In some cases, the databases do not have sufficient information to clearly delineate the applicable boundaries. The system models and generic databases were reviewed to confirm that either there was agreement between the model and generic database boundaries, or component boundaries in the current model conservatively overlap the boundaries shown in the generic databases used for the update. The failure rates for these additional components were found to be small and inclusion in the model results in, at most, a very minor conservatism in the results. The model documentation was enhanced to provide additional detail to clarify the issues with the generic database boundaries and the slightly conservative modeling approach. Regarding the example given of the battery chargers, the input and output breakers are included in the generic database boundary definition for common cause failures whereas the input breakers are not clearly identified to be included in the generic independent failure rate. The PSA model does not include common cause failure of the input or output breakers. The model does conservatively include independent failure of the input breakers due to specific modeling considerations. This approach is considered appropriate to satisfy the SR requirement.	**No significant impact** As noted, any differences in boundary definitions would at most result in a very minor conservatism and would have no significant impact on this application.

TABLE A.2-2
SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP3 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
4-14	*(continued)* 3. Sections 1.2 and 1.4 of Appendix D1 state that the data analysis package is consistent with the system analysis. However, as discussed in Item number 1 above, systems analysis only defines the system boundary and not the component boundaries within the system. 4. Boundaries of the test and maintenance unavailability events are not specifically discussed, but seem to be same as the boundaries for the failure rates. Data from the Maintenance Rule program is used for this case, but it is not clear if the system and component boundaries considered in this program is consistent with the PSA model boundaries. Section 6.3.11 of Appendix D0 discusses this issue, but there is no evidence that the analysis done in Appendix D1 considered boundaries applies to routine test and maintenance practices at IP3.				Regarding the test and maintenance boundaries, the IP3 Maintenance Rule Basis documents for each system, which define the functions the system must meet and the interfacing boundaries between systems, were compared to the maintenance unavailability terms in the updated model. The system functions are consistent with the system models. The unavailable hours monitored under the Maintenance Rule were assigned to the same major components in the model so that the model boundaries agree with or conservatively overlap the maintenance unavailability boundaries.	

TABLE A.2-2
SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP3 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
6-1	The justification/statement that the CST inventory is sufficient for AFW for 24 hrs should be enhanced.	SC-B1 SY-B11	The justification/ statement that the CST inventory is sufficient for AFW for 24 hrs should be enhanced. IP-RPT-10-00023, Rev. 0, Appendix B, Section B1.3.1.3.2 states early that CST inventory is sufficient for 24 hrs while later reveals that the MAAP analysis shows insufficient CST inventory with statement that alignment to the city water supply may be required. An informal calculation with the minimum flow requirement in EOP concludes that "it would seem that there is enough inventory in the CST to allow the AFW system to operate for 24 hours". Then in IP-RPT-10-0023, Section Insights states that 'As the normal CST inventory is sufficient to supply the AFW pumps for the 24-hour mission time in the PSA', no credit is taken for the alternate suction path from city water supply.	Perform rigorous evaluation/justification of the CST inventory to support 24-hour AFW operation.	**Resolved** Plant design documentation supports the 24 mission time for the CST. The Appendix B write-up was revised to reference a June 2004 Westinghouse calculation in support of IP3 power uprate project. The results of this calculation (along with initial calculation boundary conditions) are used to document adequate CST water inventory supply to support AFW operation for secondary-side cooling for 24 hours. In addition, as noted, CST inventory is typically maintained above the minimum inventory level, providing additional margin. Final model documentation was modified to remove the apparent discrepancies.	**No impact** Documentation issue - incorporated in final project file for the model used for this application.

TABLE A.2-2
SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP3 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
6-6	Supporting requirement IFSO-A4 is intentionally not met as stated in IP-RPT-10-00023, Rev. 0, Appendix C1, Section 3.3: 'The one supporting requirement of the ASME standard that we have made no attempt to meet is IF-B2: "for each potential source of flooding, identify the mechanisms that would result in a flooding release". In this analysis, no distinction was made between the various causes of floods because the rupture frequencies used included all floods."	IFSO-A4	This supporting requirement is intentionally not met as stated in IP-RPT-10-00023, Rev. 0, Appendix C1, Section 3.3: 'The one supporting requirement of the ASME standard that we have made no attempt to meet is IF-B2: "for each potential source of flooding, identify the mechanisms that would result in a flooding release". In this analysis, no distinction was made between the various causes of floods because the rupture frequencies used included all floods."	Identify the flooding mechanisms that would result in a release for each potential source of flooding to meet the SR.	**Resolved** The intent of the statement in the report was to acknowledge that the EPRI data used for the analysis included all rupture mechanisms that contribute to piping system failures and to note there are no readily available data that would allow us to distinguish between different release mechanisms. The identification of specific causes of failure is therefore a documentation issue. The only contributor not included in the EPRI data is human induced flooding events. Since no applicable generic data exists related to human induced events, plant specific condition reports were reviewed for applicable events (none were identified) and discussions were held with plant operations personnel. Based on those discussions, activities that could challenge system integrity such as large scale movements of water and plant modifications are typically performed during outages and would not constitute significant contributors to flooding risk. Nonetheless, the model documentation has been modified to specifically discuss both failure mechanisms and the conclusions of these human induced failure evaluations.	**No impact** As noted in the disposition, plant specific condition reports were reviewed for applicable events involving human induced flooding events, which were the only events not covered by the EPRI data. No such events were found and the frequencies used remain valid. The model documentation has been modified to specifically discuss both failure mechanisms and the conclusions of these human induced failure evaluations.
6-7	As stated in IP-RPT-10-00023, Rev. 0, Appendix C1, Table 3.3.1.1 for IFSO-A5, maximum flow rate resulting from a guillotine rupture is determined and used, instead of identifying the characteristic of release for different failure mechanism.	IFSO-A5	As stated in IP-RPT-10-00023, Rev. 0, Appendix C1, Table 3.3.1.1 for IFSO-A5, maximum flow rate resulting from a guillotine rupture is determined and used, instead of identifying the characteristic of release for different failure mechanism. This is in contrary to the SR.	Identify the characteristic of release for each source and its identified failure mechanism.	**Resolved** We consider this a documentation issue. While the table mentioned in the finding did state that a maximum flow rate resulting from a guillotine rupture was determined, it also noted that the frequency of this and lesser releases were calculated. A range of release sizes consistent with the available EPRI pipe rupture frequency data were, in fact, considered and a flow rate and frequency of occurrence derived for each. By this means, the size and frequency of possible releases were matched as required for the quantitative determination of the consequences of internal flooding. The text in the report has been modified to clarify this matter. Additional information regarding the pressures and temperatures of the ruptured systems has also been added to the documentation.	**No impact** Documentation issue - incorporated in final project file for the model used for this application.

TABLE A.2-2
SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP3 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
6-8	IP-RPT-10-00023, Rev. 0, Appendix C1, Section 4.1.3 states that the potential flood sources were identified by walkdowns and the examination of drawings, and listed in Appendix A, Plant Walkdown. However, Appendix A does not provide adequate information on flood source as (1) some flood areas are not included in the walkdown such as 3PAB41-1A, 43-60A, 46-73A, 55-63A, 3FH72-B, 3FH80-A, etc.; (2) Appendix A has stressed that the walkdown notes do NOT provide a definitive listing of all equipment and lines or other flood sources. Also other fluid sources have not been considered in the analysis.	IFSO-A1 IFSO-B1 IFSO-B2 IFSO-A3 IFSO-A6	IP-RPT-10-00023, Rev. 0, Appendix C1, Section 4.1.3 states that the potential flood sources were identified by walkdowns and the examination of drawings, and listed in Appendix A, Plant Walkdown. However, Appendix A does not provide adequate information on flood source as (1) some flood areas are not included in the walkdown such as 3PAB41-1A, 43-60A, 46-73A, 55-63A, 3FH72-B, 3FH80-A, etc.; (2) Appendix A has stressed that the walkdown notes do NOT provide a definitive listing of all equipment and lines or other flood sources. Also other fluid sources have not been considered in the analysis.	Identify the potential sources of flooding for each flood area per the standard. Perform and document walkdowns for missed flood areas. If these areas cannot be walked down for operational or health reasons, other methods of obtaining this data (e.g., plant drawings, operator interviews, etc.) should be employed and documented. Prepare an integrated list of the internal flood sources.	**Resolved** All accessible flood areas were included in the plant walkdowns. Appendix A has been revised to include the areas that were previously omitted from the documentation, including those areas mentioned in the finding. The statement in the introduction to the walkdown notes was intended only to acknowledge that there might be small bore, field run piping (less than 1 inch diameter) that were not shown on system drawings and would not have been confirmed by the walkdown. Such small bore pipes were not considered to be significant flood sources.	**No impact** Since as noted in the disposition, all areas were, in fact, walked down, this was a documentation issue and was incorporated in final project file for the model used for this application.
6-11	IP-RPT-10-00023, Rev. 0, Appendix C, Section 4.1.3, which is the section in the main report for flood sources, just refers Appendix A, Plant Walkdown for the information. There is no list of the internal flood sources in the analysis that may facilitate PRA applications, upgrades, and peer review.	IFSO-B1	There is no list of the internal flood sources in the analysis that may facilitate PRA applications, upgrades, and peer review. It could facilitate applications, update and review if sources were identified in the main report.	Prepare an integrated list of the internal flood sources.	**Resolved** This is documentation issue. A list of internal flooding sources has been developed and was included in a new Table 4.2.1.1 in the final update report. This table identifies all the flooding sources in each area and identifies adjacent or lower areas through which floodwater might propagate.	**No impact** Documentation issue - incorporated in final project file for the model used for this application.

TABLE A.2-2
SUMMARY OF INDUSTRY PEER REVIEW FINDINGS FOR THE IP3 INTERNAL EVENTS PRA MODEL UPDATE

FINDING	FINDING DESCRIPTION	ASSOC. SR	BASIS FOR PEER REVIEW FINDING	REVIEW TEAM SUGGESTED RESOLUTION	DISPOSITION	IMPACT ON ILRT APPLICATION
6-12	IP-RPT-10-00023, Rev. 0, Appendix C identifies applicable flood sources in its Appendix A, Plant Walkdown, which is not adequate for process documentation purpose. For example, the walkdown notes stressed that they do NOT provide a definitive listing of all equipment and lines or other flood sources; there is no list of sources to be examined.	IFSO-B2	IP-RPT-10-00023, Rev. 0, Appendix C identifies applicable flood sources in its Appendix A, Plant Walkdown, which is not adequate for process documentation purpose. For example, the walkdown notes stressed that they do NOT provide a definitive listing of all equipment and lines or other flood sources; there is no list of sources to be examined.	Provide adequate documentation on the process used to identify applicable flood sources	**Resolved** Although Section 3.1.2 previously described the process for identifying flooding sources, additional description has been added to that section and an additional table (Table 4.2.1.1) has been added, which provides additional detail describing the sources in each flood zone. The statement in the introduction to the walkdown notes was intended only to acknowledge that there might be small bore, field run piping (less than 1 inch diameter) that were not shown on system drawings and would not have been confirmed by the walkdown. Such small bore pipes were not considered to be significant flood sources.	**No impact** Documentation issue - incorporated in final project file for the model used for this application.

A.3 IDENTIFICATION OF KEY ASSUMPTIONS

The methodology employed in this risk assessment followed the NEI guidance. 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.4 SUMMARY

A PRA technical adequacy evaluation was performed consistent with the requirements of RG-1.200, Revision 2. This evaluation combined with the details of the results of this analysis demonstrates with reasonable assurance that the proposed extension to the ILRT interval for IP2 and IP3 to fifteen years satisfies the risk acceptance guidelines in RG 1.174.

A.5 REFERENCES

[A.1] Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities, Revision 2, March 2009.

[A.2] Engineering Report, IP2-RPT-09-00026, Rev.0, "Indian Point Unit 2 Probabilistic Safety Assessment (PSA)", November 2011.

[A.13] Engineering Report, IP3-RPT-10-00023, Rev.0, "Indian Point Unit 3 Probabilistic Safety Assessment (PSA)", November 2012.

[A.4] Entergy Fleet Procedure EN-DC-151, Revision 2, "PSA Maintenance and Update", January 2011.

[A.5] PWR Owners Group LTR-RAM-II-09-092, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Indian Point 2 Nuclear Power Plant Probabilistic Risk Assessment," March 2010.

[A.6] PWR Owners Group LTR-RAM-I-11-055, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Indian Point 3 Probabilistic Risk Assessment," October 2011.

[A.7] "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325", EPRI, Palo Alto, CA: 2008. 1018243.

[A.8] Entergy Engineering Report, IP-RPT-07-00007, "IP2 Cost Benefit Analysis of Severe Accident Mitigation Alternatives", Revision 0.

[A.9] Entergy Engineering Report, IP-RPT-07-00008, "IP3 Cost Benefit Analysis of Severe Accident Mitigation Alternatives", Revision 0.