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10 CFR 50.90

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U.S. Nuclear Regulatory Commission
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Salem Generating Station - Unit 1
Facility Operating License No. DPR-70
NRC Docket No. 50-272

Subject: License Amendment Request for One-Time Extension of the Type A Test Interval

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests an amendment to the Facility Operating License (FOL) listed above for Salem Generating Station, Unit 1. The proposed amendment revises Technical Specification 6.8.4.f, "Primary Containment Leakage Rate Testing Program" to allow a one-time extension of the Type A Integrated Leakage Rate Test (ILRT) interval for no more than five (5) years.

Attachment 1 to this letter describes the proposed changes and provides justification for the changes. PSEG has concluded that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92. Attachment 2 provides the marked up Technical Specification pages.

The proposed amendment is risk-informed and follows the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Bases." PSEG performed a plant-specific evaluation which demonstrates that the increase in risk resulting from the proposed amendment is small and within established guidance. A copy of the risk assessment is provided in Attachment 3.

PSEG requests NRC approval of the proposed License Amendment by September 30, 2010. Once approved, the amendment shall be implemented within 60 days. Approval by the requested date will support planning activities for Salem Unit 1 refueling outage 1R21, currently scheduled to begin in Fall 2011.

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The proposed changes have been reviewed by the Plant Operations Review Committee. There are no commitments contained in this letter.

We are notifying the State of New Jersey of this application for changes to the TS and Operating License by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Mr. Paul Duke at 856-339-1466.

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

Executed on 9/21/09
(date)

Sincerely,

Robert C. Braun
Site Vice President
Salem Generating Station

Attachments (3)
1. Description of Proposed Changes, Technical Analysis, and Regulatory Analysis
2. Markup of Technical Specification page
3. Risk Assessment to Support ILRT (Type A) Interval Extension Request

cc: S. Collins, Regional Administrator – NRC Region I
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector - Salem
P. Mulligan, Manager IV, NJBNE
L. Marabella, Corporate Commitment Tracking Coordinator
H. Berrick, Station Commitment Tracking Coordinator

ATTACHMENT 1

License Amendment Request

Salem Generating Station - Unit 1
NRC Docket No. 50-272

Description of Proposed Changes, Technical Analysis, and Regulatory Analysis

Subject: One-Time Extension of the Type A Test Interval

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1.0 DESCRIPTION

This letter is a request to amend Facility Operating License DPR-70 for the Salem Nuclear Generating Station, Unit 1 (SGS-1). The proposed change would revise Technical Specification (TS) 6.8.4.f, "Primary Containment Leakage Rate Testing Program," to permit a one-time extension of the containment Type A Integrated Leak Rate Test (ILRT) interval from ten to fifteen years.

PSEG Nuclear LLC (PSEG) requests NRC approval of the proposed License Amendment by September 30, 2010. Once approved, the amendment shall be implemented within 60 days.

2.0 PROPOSED CHANGE

The proposed change would revise TS 6.8.4.f, "Primary Containment Leakage Rate Testing Program," to add the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after May 7, 2001, shall be performed no later than May 7, 2016.

The marked up TS page is provided in Attachment 2.

3.0 BACKGROUND

<u>Primary Containment</u>
The reactor containment structure is a reinforced concrete vertical right cylinder with a flat base and a hemispherical dome. A welded steel liner with a minimum thickness of 1/4 inch is attached to the inside face of the concrete shell to ensure a high degree of leak tightness. The structure consists of side walls measuring 142 feet in height from the liner on the base to the springline of the dome, and has an inside diameter of 140 feet. The side walls of the cylinder and the dome are 4 feet-6 inches and 3 feet-6 inches thick, respectively. The inside radius of the dome is equal to the inside radius of the cylinder so that the discontinuity at the springline due to the change in thickness is on the outer surface. The flat concrete base mat is 16 feet thick with a bottom liner plate located on top of this mat. The base mat liner is connected to the cylinder liner with a 3/4 inch knuckle plate. The majority of the lower cylinder wall sections are insulated to prevent buckling of the liner due to restricted growth under a temperature rise. The underground portion of the containment structure is waterproofed with an impervious membrane that prevents seepage of ground water through cracks in the concrete.

The design objective of the containment structure is to contain all radioactive material which might be released from the reactor core following a design basis loss-of-coolant accident (LOCA). The containment structure serves as both a biological shield and a pressure container. The inside surface of the liner plate in the cylinder and dome is painted with catalyzed epoxy paint. Numerous penetrations pass through the concrete structure including piping and electrical penetrations and hatches.

The containment structure is designed to limit post accident leakage to 0.1 percent of the total containment free air volume per day (La) when pressurized to the design accident pressure of 47.0 psig (Pa). The completed containment structure was subjected to a one time 54 psig air pressure test (115% of the design pressure) to verify structural integrity. This pressure was maintained for a minimum period of 1 hour while measurements and observations of the structure were conducted.

Periodic containment leakage tests are performed on the containment at frequencies specified in plant TS. The results of those leakage tests are provided in the Table in Section 4.0.

10 CFR 50 Appendix J Test Requirements

10 CFR 50.54(o) and 10 CFR 50 Appendix J, require periodic tests to assure that leakage through the primary reactor containment and systems and components penetrating primary containment does not exceed allowable leakage rate values as specified in the TS. Appendix J requires three types of tests: (1) Type A tests, intended to measure the primary reactor containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage-limiting boundaries for primary containment penetrations other than valves; and (3) Type C tests, intended to measure containment isolation valve leakage rates.

TS 6.8.4.f requires that a program be established to comply with the containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The program is required to be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. RG 1.163 endorses, with certain exceptions, NEI 94-01, Revision 0, as an acceptable method for complying with the provisions of Appendix J, Option B.

RG 1.163, Section C.1 states that licensees intending to comply with the 10 CFR 50, Appendix J, Option B should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 rather than using test intervals specified in ANSI/ANS-56.8-1994. NEI 94-01, Section 11.0, refers to Section 9, which states that Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 La. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

As noted in NEI 94-01, most containment leakage is identified by local leakage rate testing (Type B and C tests). The purpose of Type A testing is to verify the leakage integrity of the containment structure. The primary performance objective of the Type A test is not to quantify an overall containment system leakage rate.

Type A Test Interval Extensions

NUREG-1493, "Performance-Based Containment Leak-Test Program," concluded in Section 10.1.2, that reducing the frequency of Type A tests (ILRTs) from the original 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, NUREG-1493 concluded that increasing the interval between ILRTs is possible with minimal impact on public risk.

Benefits of Interval Extension for Salem Unit 1

Salem Unit 1 Type A tests are currently performed once per 10 years, based on successful performance history. The current 10-year interval for the completion of the next Type A test ends on May 7, 2011. PSEG plans to perform the next Salem Unit 1 Type A test during the Fall 2011 refueling outage, using a portion of the 15-month extension provision of NEI 94-01, Revision 0. The proposed change would allow the next Type A test to be performed no later than May 7, 2016. This one-time extension would provide substantial benefits in the form of reduced personnel exposure and reduced outage costs.

4.0 TECHNICAL ANALYSIS

Type A Test History

The completed containment structure was subjected to a one time 54 psig air pressure test (115% of the design pressure) to verify structural integrity. This pressure was maintained for a minimum period of 1 hour while measurements and observations of the structure were conducted. Periodic containment leakage tests are performed on the containment at frequencies specified by plant TS. The results of those leakage tests are provided in the Table below. All tests passed the as-found acceptance criteria of 1.0 La, where La is the maximum allowable leakage rate at pressure Pa.

Salem Unit 1 Type A (ILRT) Test Results

Test Date	Leakage Expressed as a Fraction of 1.0 La
12/87	0.63 La
4/91	0.417 La
5/01	0.095 La

Type A (ILRT) test results are influenced by a number of factors. Due to these various factors the results are not trendable and can vary significantly from test to test. Some of contributions to the results obtained are as follows:

- Implementation of Performance Based testing under Option B to 10 CFR 50 Appendix J has changed the basic test set-up. Prior to Option B, systems that penetrated containment were extensively drained, vented and aligned as part of the test boundary. Under Option B, the Type A test is used to measure the leakage from the containment structure itself and not individual penetration leakage paths tested under the Type B and C program. Type B and C measured leakages however are calculated into the final Type A test results

- There is a strong station effort to maintain penetration leakage as low as possible through improved maintenance practices. Individual component leakage limits have been revised to reflect expected component performance rather than penetration line size. Station outage goals are established for maintaining overall penetration Type B and C leakage rates.

- Test methods for pressurizing containment have improved helping to achieve a more stable environment over a shorter period of time.

Type B and C Test History

The Type B and C maximum penetration pathway as-left leakage rates for the last six Salem Unit 1 outages are shown in the table below. The values are expressed in standard cubic centimeters per minute (sccm). At Salem the value of La is 216,250 sccm. The TS allowable maximum pathway total Type B and C leakage is 0.6 La, which equals approximately 129,750 sccm.

Refuel Outage No.	Date	As-left Leakage (sccm)
1R14	Spring 2001	24,515.0
1R15	Fall 2002	42,176.4
1R16	Spring 2004	28,490.1
1R17	Fall 2005	34,043.6
1R18	Spring 2007	27,505.1
1R19	Fall 2008	26,011.1

Schedule of Type B and C Tests during Extended Interval

The following table identifies the current Type B and C penetration test frequencies. The test frequencies are established based on performance utilizing the requirements of Option B. The test frequencies are re-evaluated after each refueling outage for potential changes. Also attached are the refueling outages in which these tests are currently planned. The schedule presumes that the ILRT will be performed in refueling outage 1R24 (Spring 2016). The dates for the ILRT and Type B and C tests are subject to change.

It should be noted that some of the scheduled test dates will be modified to ensure that penetrations and components not drained and vented during the next scheduled ILRT have current test results within the previous 24 month period as required by NEI 94-01, section 9.2.1.

Type B Tests								
Penetration	**Frequency**	**Last Test Date**	**1R20 Spring 2010**	**1R21 Fall 2011**	**1R22 Spring 2013**	**1R23 Fall 2014**	**1R24 Spring 2016**
Elect Pen 1-1	10 YR	10/11/2005				X	
Elect Pen 1-2	10 YR	10/11/2005				X	
Elect Pen 1-3	10 YR	04/02/2004			X		
Elect Pen 1-4	10 YR	10/10/2005				X	
Elect Pen 1-5	10 YR	10/08/2002		X			
Elect Pen 1-6	10 YR	04/05/2001	X				
Elect Pen 1-7	10 YR	10/10/2005				X	
Elect Pen 1-8	10 YR	10/08/2002		X			
Elect Pen 1-9*	10 YR	10/09/2008					X
Elect Pen 1-10	10 YR	10/08/2002		X			
Elect Pen 1-11	10 YR	10/08/2002		X			
Elect Pen 1-12	10 YR	04/05/2001	X				
Elect Pen 1-13	10 YR	03/30/2004			X		
Elect Pen 1-14*	10 YR	10/10/2008					X
Elect Pen 1-15	10 YR	04/05/2001	X				
Elect Pen 1-16	10 YR	04/02/2004			X		
Elect Pen 1-17	10 YR	04/03/2001	X				
Elect Pen 1-18	10 YR	04/06/2001	X				
Elect Pen 1-19*	10 YR	10/15/2008					X
Elect Pen 1-20	10 YR	10/07/2005				X	
Elect Pen 1-21*	10 YR	10/10/2008					X
Elect Pen 1-23	10 YR	10/11/2005				X	
Elect Pen 1-24	10 YR	10/07/2005				X	
Elect Pen 1-25	10 YR	04/02/2004			X		
Elect Pen 1-26	10 YR	10/10/2005				X	
Elect Pen 1-27	10 YR	10/08/2002		X			
Elect Pen 1-28	10 YR	04/05/2001	X				
Elect Pen 1-29	10 YR	10/10/2005				X	
Elect Pen 1-31	30 M	10/09/2008	X	X	X	X	X
Elect Pen 1-32	10 YR	10/08/2002		X			
Elect Pen 1-33	10 YR	10/08/2002		X			
Elect Pen 1-34	10 YR	04/05/2001	X				

Type B Tests							
Penetration	Frequency	Last Test Date	1R20 Spring 2010	1R21 Fall 2011	1R22 Spring 2013	1R23 Fall 2014	1R24 Spring 2016
Elect Pen 1-35	10 YR	04/02/2004			X		
Elect Pen 1-37	10 YR	03/30/2004			X		
Elect Pen 1-38	10 YR	03/30/2004			X		
Elect Pen 1-39	10 YR	04/03/2001	X				
Elect Pen 1-40	10 YR	04/05/2001	X				
Elect Pen 1-41	10 YR	04/02/2001	X				
Elect Pen 1-42	10 YR	10/07/2005				X	
Elect Pen 1-43*	10 YR	10/10/2008					X
Elect Pen 1-45	10 YR	04/02/2004			X		
Elect Pen 1-46	10 YR	10/10/2005				X	
Elect Pen 1-47*	10 YR	10/10/2008					X
Elect Pen 1-48	10 YR	10/10/2005				X	
Elect Pen 1-49	10 YR	04/05/2001	X				
Elect Pen 1-50	10 YR	10/08/2002		X			
Elect Pen 1-52	10 YR	04/02/2004			X		
Elect Pen 1-53	10 YR	04/02/2004			X		
Elect Pen 1-56	10 YR	10/08/2002		X			
Elect Pen 1-57*	10 YR	10/09/2008					X
Spare Mech 1-58	10 YR	10/05/2005				X	
Elect Pen 1-59	10 YR	04/02/2004			X		
Elect Pen 1-60	10 YR	04/02/2004			X		
Elect Pen 1-61	10 YR	04/03/2001	X				
Elect Pen 1-62*	10 YR	10/10/2008					X
Elect Pen 1-63	10 YR	10/08/2002		X			
Elect Pen 1-64*	10 YR	10/10/2008					X
Elect Pen 1-65*	10 YR	10/10/2008					X
100' A/L E-Pen L	10 YR	03/22/2007					X
100' A/L E-Pen R	10 YR	03/22/2007					X
130' A/L E-Pen L	10 YR	03/22/2007					X
130' A/L E-Pen R	10 YR	03/22/2007					X
11SJ44 E-Pen	10 YR	03/22/2007					X
12SJ44 E-Pen	10 YR	03/22/2007					X
11SJ44 Hatch*	10 YR	11/01/2008					X

Type B Tests							
Penetration	**Frequency**	**Last Test Date**	**1R20 Spring 2010**	**1R21 Fall 2011**	**1R22 Spring 2013**	**1R23 Fall 2014**	**1R24 Spring 2016**
12SJ44 Hatch*	10 YR	11/03/2008					X
100' Airlock**	30 M	07/22/2009					
130' Airlock**	30 M	07/21/2009					
Refuel Flange*	10 YR	11/03/2008					X
Equip Hatch*	10 YR	11/04/2008		X			X
1SA591 Flange	30 M	10/12/2008	X	X	X	X	X
1VC2 Blind Flange	30 M	11/04/2008	X	X	X	X	X
1VC3 Blind Flange	30 M	11/04/2008	X	X	X	X	X

* Next scheduled test would be 1R25, Fall, 2017. Currently scheduled in 1R24 at end of interval to meet requirement to test within 24 months of next Type A test, pending approval of proposed one-time extension.

** The airlocks are currently tested online.

Type C Valves							
Valve ID	**Frequency**	**Last Test Date**	**1R20 Spring 2010**	**1R21 Fall 2011**	**1R22 Spring 2013**	**1R23 Fall 2014**	**1R24 Spring 2016**
11CA330	30 M	10/21/2008	X	X	X	X	X
11CA360	30 M	10/21/2008	X	X	X	X	X
12CA330	30 M	10/26/2008	X	X	X	X	X
12CA360	30 M	10/26/2008	X	X	X	X	X
1CA1714*	5YR	01/10/2005					
1CA1715*	5YR	01/10/2005					
1CC118	30 M	10/20/2008	X	X	X	X	X
1CC119	30 M	10/20/2008	X	X	X	X	X
1CC136	30 M	10/20/2008	X	X	X	X	X
1CC186/187	30 M	10/20/2008	X	X	X	X	X
1CC131	30 M	10/19/2008	X	X	X	X	X
1CC190/208	30 M	10/19/2008	X	X	X	X	X
1CC113/215	5 YR	10/19/2008			X		X
11CS2/10	30 M	10/23/2008	X	X	X	X	X
11CS48	30 M	10/31/2008	X	X	X	X	X
12CS2/10	30 M	10/22/2008	X	X	X	X	X
12CS48	30 M	10/30/2008	X	X	X	X	X
1CS900/901	5YR	02/07/2006	X			X	
1CS902	5YR	02/07/2006	X			X	
1CV3/4/5	30 M	10/22/2008	X	X	X	X	X
1CV7	30 M	10/22/2008	X	X	X	X	X
11CV99	30 M	10/22/2008	X	X	X	X	X
12CV99	30 M	10/22/2008	X	X	X	X	X
13CV99	30 M	10/22/2008	X	X	X	X	X
14CV99	30 M	10/23/2008	X	X	X	X	X
1CV68	5 YR	04/05/2007	X			X	
1CV69	5 YR	04/07/2007	X			X	
1CV74	30 M	10/24/2008	X	X	X	X	X
1CV116	30 M	10/20/2008	X	X	X	X	X
1CV284/296	30 M	10/20/2008	X	X	X	X	X
1DR29	30 M	10/15/2008	X	X	X	X	X
1DR30	30 M	10/15/2008	X	X	X	X	X

Type C Valves							
Valve ID	Frequency	Last Test Date	1R20 Spring 2010	1R21 Fall 2011	1R22 Spring 2013	1R23 Fall 2014	1R24 Spring 2016
1FP147	30 M	10/17/2008	X	X	X	X	X
1FP148	30 M	10/17/2008	X	X	X	X	X
1NT25	30 M	10/21/2008	X	X	X	X	X
1NT26	30 M	10/21/2008	X	X	X	X	X
1NT32	30 M	10/19/2008	X	X	X	X	X
1NT34	30 M	10/29/2008	X	X	X	X	X
1PR17	30 M	10/20/2008	X	X	X	X	X
1PR18	30 M	10/20/2008	X	X	X	X	X
1PR25	30 M	10/25/2008	X	X	X	X	X
1SA262	5 YR	03/30/2006	X			X	
1SA264	5 YR	03/30/2006	X			X	
1SA265	5 YR	03/30/2006	X			X	
1SA267	5 YR	03/30/2006	X			X	
1SA268	5 YR	03/30/2006	X			X	
1SA270	5 YR	03/30/2006	X			X	
1SF22/WL191	5 YR	03/23/2007		X			X
1SF36/WL190	5 YR	10/12/2005	X			X	
1SJ123	5 YR	10/20/2008		X		X	
1SJ53/60	5 YR	10/20/2008		X		X	
1SS104	30 M	10/17/2008	X	X	X	X	X
1SS33	30 M	10/31/2008	X	X	X	X	X
1SS103	30 M	10/30/2008	X	X	X	X	X
1SS27/653	30 M	10/31/2008	X	X	X	X	X
1SS107	30 M	10/30/2008	X	X	X	X	X
1SS49	30 M	10/16/2008	X	X	X	X	X
1SS110	30 M	11/01/2008	X	X	X	X	X
1SS64	30 M	11/01/2008	X	X	X	X	X
1VC5/6	30 M	10/15/2008	X	X	X	X	X
1VC7	5 YR	08/07/2008		X			X
1VC8	5 YR	05/23/2008		X			X
1VC9	30 M	10/18/2008	X	X	X	X	X
1VC10	30 M	10/18/2008	X	X	X	X	X
1VC11	5 YR	03/19/2009		X			X

Type C Valves							
Valve ID	**Frequency**	**Last Test Date**	**1R20 Spring 2010**	**1R21 Fall 2011**	**1R22 Spring 2013**	**1R23 Fall 2014**	**1R24 Spring 2016**
1VC12	5 YR	04/12/2007		X			X
1VC13	5 YR	10/19/2005	X			X	
1VC14	5 YR	10/19/2005	X			X	
1WL12	5 YR	10/28/2005	X			X	
1WL13/476	5 YR	10/28/2005	X			X	
1WL16	30 M	10/25/2008	X	X	X	X	X
1WL17/478	30 M	10/25/2008	X	X	X	X	X
1WL96	5 YR	10/25/2005	X			X	
1WL97	5 YR	10/25/2005	X			X	
1WL98	5 YR	10/29/2005	X			X	
1WL99/108	5 YR	10/29/2005	X			X	
1WR80	30 M	10/16/2008	X	X	X	X	X
1WR81	30 M	10/16/2008	X	X	X	X	X

* These valves are currently tested on-line.

Containment ISI Requirements and History
IWE examinations were performed in the first containment inservice inspection (CISI) interval in accordance with the 1998 Edition, 1998 Addenda of the ASME Section XI Code and as modified by 10CFR50.55a and Relief Request No. RR-E1 (Reference 9). The most recent examinations were performed during 1R18 (Spring 2007). The second CISI interval starts in April 2010 with examinations to be performed in accordance with the 2004 Edition of the ASME Section XI Code and as modified by 10CFR50.55a. The next scheduled examinations are during 1R21 (Fall 2011) and 1R23 (Fall 2014). The last examination scheduled during the second CISI interval will be 1R26 (Spring 2019); this last exam will be beyond the requested extended period.

The first IWE examination was performed in spring 2001 (1R14) and resulted in no reportable indications. Several Notifications were processed to document examination indications of coating degradation on containment penetrations and areas on the metal containment liner during this examination campaign. Engineering evaluations were performed on noted areas of degradation and all areas were found acceptable. Corrective maintenance orders were generated to restore the degraded coatings to original configuration. A number of broken or missing liner insulation retaining studs were also identified. An evaluation determined that the missing studs did not adversely affect the structural integrity of the containment liner and had no affect on moisture intrusion for the liner.

The second IWE examination was performed in spring 2004 (1R16) and resulted in no reportable indications. Several Notifications were processed to document examination indications of coating degradation and blistering on containment penetrations and on the metal containment liner during this examination campaign. Engineering evaluations were performed on noted areas of degradation and all areas were found acceptable. Corrective maintenance orders were generated to restore the degraded coatings to original configuration.

The third and most recent IWE examination was performed in spring 2007 (1R18). No reportable or recordable indications were identified.

IWL concrete containment examinations were performed in the first CISI interval in accordance with the 1998 Edition, 1998 Addenda of the ASME Section XI Code and as modified by 10CFR50.55a and Relief Request No. RR-L1 (Reference 9). The most recent examination was performed during 1R17 (Fall 2005). The next scheduled IWL examination during the first CISI interval is scheduled during 1R20 (Spring 2010). The second CISI interval starts in April 2010 with examinations to be performed in accordance with the 2004 Edition of the ASME Section XI Code and as modified by 10CFR50.55a. Second CISI interval IWL examinations are scheduled during 1R23 (Fall 2014) and 1R26 (Spring 2019).

The first IWL examination was performed in spring 2001 (1R14) resulting in no reportable or recordable indications.

The second and most recent IWL examination performed during 1R17 (Fall 2005) resulted in no reportable indications being identified. Examination revealed some acceptable minor surface scaling and identified moisture/intrusion barrier plate bolt coating having light to medium rust. Restoration of bolt coating is planned for 1R20 (Spring 2010).

To support license renewal activities, during 1R19 (Fall 2008), insulation panels were removed to permit the liner to be inspected in four normally inaccessible areas. No rejectable areas were identified. Light rusting was observed in two of the areas, but did not require any further action. All four locations examined were considered satisfactory for continued operation.

Coating Inspection Program
PSEG has implemented controls for the procurement, application, and maintenance of Service Level I protective coatings used in containment, consistent with the licensing basis and regulatory requirements applicable to the Salem Station. The containment coatings monitoring program is based on the guidance of ASTM D5163. Defects observed during periodic visual examinations are documented in the PSEG corrective action program, assessed, and repaired or replaced as necessary.

Schedule and Method for Appendix J Visual Examination (RG 1.163, Regulatory Position C.3)
10 CFR 50 Appendix J, Option B, requires general visual inspections of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be conducted prior to each test, and at a periodic interval between tests based on the performance of the containment. RG 1.163 states that the visual examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years. Qualification of examination personnel is not specified in any of the above documents.

The most recent visual examination was completed satisfactorily in 1R18 (Spring 2007). Scheduled visual examinations during the extended interval are listed below:

Outage	Schedule
1R20	Spring, 2010
1R22	Spring, 2013
1R24	Spring, 2016

1R24 marks the end of the requested 5 year interval extension period. If the ILRT is performed during a non-scheduled outage, the visual examination would be performed in that same outage prior to the test as required and per the prerequisites of the ILRT test procedure.

As stated previously the documents specifying exam performance do not require any specific qualifications, however at Salem the examinations are typically performed by ASME Code certified visual examiners. Liner inspections are typically performed by personnel with VT-1 and VT-3 certifications and concrete inspections by examiners with VT-1C and VT-3C certifications.

Examinations are conducted in accordance with approved station procedure SH.RA-ST.ZZ-0106, Visual Inspection of Containment Structural Integrity. This procedure allows credit to be taken for examination areas which coincide with examinations performed to satisfy ASME Section XI IWE and IWL examinations performed during the same outage.

Summary of Risk Assessment
As discussed in Attachment 3, the Probabilistic Safety Risk Assessment results demonstrate a small impact in risk associated with the one-time extension of the containment Type A ILRT from 10 to 15 years. The risk assessment follows the guidelines from NEI 94-01 (Reference 2), the methodology used in EPRI TR-104285 (Reference 6), the NEI Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment

Integrated Leakage Rate Test Surveillance Intervals (Reference 7), 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 (Reference 3), and the methodology used for Calvert Cliffs (Reference 10) to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval. 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 (Reference 11). An assessment of the technical adequacy of the PRA model used for the risk assessment following the guidance provided in Regulatory Guide 1.200 is included in Appendix A to Attachment 3.

The following is a brief summary of some of the key aspects of the Type A ILRT interval extension risk assessment to 15 years:

- The acceptance guidelines in RG 1.174 are used to assess the acceptability of this onetime extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF for Salem Unit 1, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year provided that the total from all contributors (including external events) can be reasonably shown to be less than 10^{-5} per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also calculated. In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter based on the precedent set by previous submittals for ILRT extensions. (No criteria have been established for this parameter change.)

- The increase in internal events LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is estimated as 4.06E-07/yr (i.e. in the "small" change region using the acceptance guidelines of RG 1.174) using the NEI guidance as written, and at 4.36E-08/yr (i.e. in the "very small" change region) using the EPRI Expert Elicitation methodology. The increase in internal events LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years for the base case with corrosion included is 4.22E-07/yr which also falls in the "small" change region of the acceptance guidelines in RG 1.174.

- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen year interval is about 0.83% using the NEI guidance, and drops to about 0.09% using the EPRI Expert Elicitation methodology. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.

- The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by

Type A testing, is 7.91E-01 person-rem/yr (1.1% increase from base dose) using the NEI guidance, and drops to 1.51E-01 person-rem/yr (0.2% increase from base dose) using the EPRI Expert Elicitation methodology. Therefore, in either case, the risk impact when compared to other severe accident risks is negligible.

- To determine the potential impact from external events, an additional bounding assessment from the risk associated with external events utilizing the information from the Salem IPEEE was performed. The total increase in LERF due to internal events and the bounding external events assessment is 8.44E-07/yr, which is in Region II (i.e. "small change") of the RG 1.174 acceptance guidelines.
- Finally, the same bounding analysis indicates that the total LERF from internal and external risks is 8.19E-06/yr, which is less than the RG 1.174 limit of 10^{-5}/yr given that the ΔLERF is in Region II.

Therefore, increasing the ILRT interval to 15 years is not considered to be significant since it represents a small change to the Salem Unit 1 risk profile.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

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

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

   Response: No.
   The proposed change would revise Technical Specification (TS) 6.8.4.f, "Primary Containment Leakage Rate Testing Program" to permit a one-time extension of the containment Type A Integrated Leak Rate Test (ILRT) from ten to fifteen years.

   The function of the containment is to isolate and contain fission products released from the reactor coolant system following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated with the performance of containment leakage testing is not an initiating event for any accident previously evaluated. There are no physical changes being made to the containment structure and no change made to the containment allowable leakage rate specified in Technical Specifications.

   During the extended test interval, containment integrity will continue to be assured by programs for local leak rate testing and containment inspections are routinely performed as required by ASME Code which demonstrates the structural integrity of the primary containment. The proposed changes do not affect performance of the containment, reactor operations or accident analysis.

The risk assessment of the proposed change has concluded that there is not a significant increase in the consequences of an accident as measured by the Large Early Release Frequency, Population Dose, and Conditional Containment Failure Frequency. These results show that an ILRT test extension will not represent a significant increase in the consequences of an accident.

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 change for a one-time, five-year extension of the Type A test makes no physical changes to the plant or to plant operations. No credible new failure mechanisms, malfunctions or accident initiators are being introduced by the proposed change.

   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?

   Response: No.
   The integrity of the containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the containment is verified by a Type A ILRT, as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The proposed change does not affect the method or acceptance criteria for Type A, B and C testing. During the extended test interval, containment inspections performed in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection," and 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," provide assurance that the containment will not degrade in a manner that is only detectable by Type A testing.

   The effect of the proposed change on Large Early Release Frequency, person-rem, and Conditional Containment Failure Frequency was determined not to be significant.

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

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

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix J, Option B, Section V.B.3 requires that the regulatory guide or other implementation document used to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.

The proposed change will revise TS 6.8.4.f to reflect a one-time extension to the Salem Unit 1 Type A ILRT as currently specified in the Technical Specifications. The one-time extension deviates from the guidelines contained in Regulatory Guide (RG) 1.163 and NEI 94-01, Rev. 0. The proposed change is consistent with the requirements of 10 CFR 50.36(c)(5) and 10 CFR 50, Appendix J, Option B, Section V.B.3. Therefore this change does not require an exemption from 10 CFR 50, Appendix J, Option B.

6. ENVIRONMENTAL CONSIDERATION

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

7. PRECEDENT

The NRC has approved similar risk-informed license amendment requests relating to a one-time extension of the ILRT interval for a number of plants. In addition to Salem Unit 2 (ADAMS Accession No. ML021300259), recent examples include:

- Braidwood Station, Units 1 and 2 (ADAMS Accession No. ML080640290)
- Point Beach Nuclear Plant, Units 1 and 2 (ADAMS Accession No. ML080380356)
- Byron Station, Unit Nos. 1 and 2 (ADAMS Accession No. ML080350348)

The precedent examples apply to the proposed interval extension for Salem Unit 1 based on containment type (PWR dry atmospheric); conduct of Type A, Type B, and Type C testing in accordance with 10 CFR 50, Appendix J, Option B; containment inservice inspection programs in accordance with Subsections IWE and IWL of Section XI of the ASME Boiler and Pressure Vessel Code; and risk assessments following the guidelines from NEI 94-01, the methodology used in EPRI TR-104285, the NEI Interim Guidance, and the NRC regulatory guidance in RG 1.174.

8. REFERENCES

1. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995 (RG 1.163).
2. NEI 94-01, "Nuclear Energy Institute Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995 (NEI 94-01).
3. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.
4. American Society of Mechanical Engineers (ASME) Section XI Subsections IWE and IWL, (Reactor Building Containment Inspections).
5. NUREG-1493, "Performance-Based Containment Leak-Test Program," Final Report, September 1995 (NUREG-1493).
6. Electric Power Research Institute, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals", EPRI TR-104285, August 1994.
7. Letter from A. Pietrangelo (NEI) to NEI Administrative Points of Contact, Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals, November 13, 2001.
8. NEI Memo to the USNRC, "One-time extensions of containment integrated leak rate test interval -additional information," November 30, 2001.
9. Hope Creek Nuclear Generating Station and Salem Nuclear Generating Station, Units 1 and 2 - Evaluation of Relief Requests: Use of 1998 Edition of Subsections IWE and IWL of the ASME Section XI Code for Containment Inspections (TAC Nos. MA6865, MA6866, and MA6867)
10. "Calvert Cliffs Nuclear Power Plant Unit No. 1; Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, "Constellation Nuclear letter to USNRC, March 27, 2002.
11. Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325. EPRI, Palo Alto, CA: October 2008. 1018243.

ATTACHMENT 2

Salem Generating Station, Unit 1

Facility Operating License No. DPR-70
NRC Docket No. 50-272

One-Time Extension of the Type A Test Interval

Markup of Proposed Technical Specification Page Changes

Revised TS Pages
6-19

Insert A

, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after May 7, 2001, shall be performed no later than May 7, 2016.

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. <u>Backup Method for Determining Subcooling Margin</u>

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling Margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring

e. Deleted

6.8.4.f. <u>Primary Containment Leakage Rate Testing Program</u>

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10CFR50.54(o) and 10CFR50, 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 or INSERT A

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 47.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup

ATTACHMENT 3

Salem Generating Station, Unit 1

Facility Operating License No. DPR-70
NRC Docket No. 50-272

One-Time Extension of the Type A Test Interval

Risk Assessment for Salem Unit 1 to Support ILRT (Type A) Interval Extension Request

RM DOCUMENTATION NO:	S1-LAR-01	REV: 0	PAGE NO. i

STATION: Salem Generating Station

UNIT(S) AFFECTED: 1

TITLE: Risk Assessment to Support ILRT (Type A) Interval Extension Request

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

[ ] Review required after periodic Update

[X] Internal RM Documentation **[ ] External RM Documentation**

Electronic Calculation Data Files:

Microsoft Excel File "Salem_ILRT-07132009.xls", 7/13/2009, 12:23 PM, 208 KB

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

This RM documentation supersedes: N/A **in its entirety.**

CHRISTOPHER PUPEK

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1.0 PURPOSE OF ANALYSIS

1.1 PURPOSE

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Salem Generating Station, Unit 1 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 used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3, 21], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [19]. The format of this document is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the October 2008 EPRI final report [22].

1.2 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than normal containment leakage of 1.0La (allowable leakage).

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [5], "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285 [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 that for a representative PWR plant (i.e. Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for Salem Unit 1.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In November and December 2001, NEI issued enhanced guidance (hereafter referred to as the NEI Interim Guidance) that builds on the TR-104285 methodology and intended to provide for more consistent submittals [3, 21]. The NEI Interim Guidance was developed for NEI by EPRI using personnel who also developed the TR-104285 methodology. This ILRT interval extension risk assessment for Salem Unit 1 employs the NEI Interim Guidance methodology, with the affected System, Structure, or Component (SSC) being the primary containment boundary.

1.3 ACCEPTANCE CRITERIA

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF for Salem Unit 1, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year provided that the total from all contributors (including external events) can be reasonably shown to be less than 10^{-5} per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also calculated.

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

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years [22]. The approach is consistent with that presented in NEI Interim Guidance [3, 21], EPRI TR-104285 [2], NUREG-1493 [5] and the Calvert Cliffs liner corrosion analysis [19]. The analysis uses results from a Level 2 analysis of core damage scenarios from the current Salem Unit 1 PRA model and the subsequent containment responses for the various fission product release categories including no or negligible release.

The six general steps of this assessment are as follows:

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

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

- Consistent with the other industry containment leak risk assessments, the Salem Unit 1 assessment uses population dose as one of the risk measures. The other risk measures used in the Salem Unit 1 assessment are LERF and the conditional containment failure probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.

- This evaluation for Salem Unit 1 uses ground rules and methods to calculate changes in risk metrics that are similar to those used in the EPRI approach.

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The Salem Unit 1 Level 1 and Level 2 internal events PRA models provide representative results.
- It is appropriate to use the Salem Unit 1 internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations; however fire and seismic events have been accounted for in the analysis based on the available information from the Salem IPEEE [18] as described in Section 5.7.
- Dose results for the containment failures modeled in the PRA can be characterized by information provided in Salem SAMA to be submitted to NRC in 2009 [9] using the 1990 and 2000 census population reports.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is 10La, based on the previously approved methodology performed for Indian Point Unit 3 [6, 7].
- The representative containment leakage for Class 3b sequences is 100La, based on the recommendations in the latest EPRI report [22]. Note that most of the previous ILRT extension requests utilized 35La.
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [6, 7]. The Class 3b category increase is used as a surrogate for LERF in this application even though the releases associated with a 100La release would not necessarily be consistent with a "Large" release for Salem Unit 1.
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes to the conclusions from this analysis will result from this separate categorization.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The use of estimated 2015 population data is adequate for this analysis. Precise evaluations of the projected population would not significantly impact the quantitative results, nor would it change the conclusions.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8].

4.0 INPUTS

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

4.1 GENERAL RESOURCES AVAILABLE

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

1. NUREG/CR-3539 [10]
2. NUREG/CR-4220 [11]
3. NUREG-1273 [12]
4. NUREG/CR-4330 [13]
5. EPRI TR-105189 [8]
6. NUREG-1493 [5]
7. EPRI TR-104285 [2]
8. NEI Interim Guidance [3, 21]
9. Calvert Cliffs liner corrosion analysis [19]
10. EPRI 1018243 [22]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eighth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The ninth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the last study complements the previous EPRI report [2], integrates the NEI interim guidance, and provides the results of an expert elicitation process to determine the relationship between pre-existing containment leakage probability and magnitude.

NUREG/CR-3539 [10]

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

NUREG/CR-4220 [11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories 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 [12]

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

NUREG/CR-4330 [13]

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

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

EPRI TR-105189 [8]

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

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

NUREG-1493 [5]

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

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

- 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 [14] Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

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

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

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

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

Release Category Definitions

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

TABLE 4.1-1
EPRI/NEI CONTAINMENT FAILURE CLASSIFICATIONS [2]

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

NEI Interim Guidance [3, 21]

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

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

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

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

The approach included in this guidance document is used in the Salem Unit 1 assessment to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI Class 3a and 3b scenarios in this analysis as described in Section 5.

<u>Calvert Cliffs Liner Corrosion Analysis [19]</u>

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

<u>EPRI 1018243 [22]</u>

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

The risk impact assessment using the Jeffrey's Non-Informative Prior statistical method is further supplemented with a sensitivity case using expert elicitation performed to address conservatisms. The expert elicitation is used to determine the relationship between pre-existing containment leakage probability and magnitude. The results of the expert elicitation process from this report are used as a separate sensitivity investigation for the Salem Unit 1 analysis presented here in Section 6.2.

4.2 PLANT-SPECIFIC INPUTS

The Salem Unit 1 specific information used to perform this ILRT interval extension risk assessment includes the following:

- Level 1 Model results [16]
- Level 2 Model results [17]
- Population within a 50-mile radius [9]

<u>Salem Unit 1 Internal Events Level 1 PRA Model</u>

The current Level 1 PRA model is an event tree / linked fault tree model characteristic of the as-built, as-operated plant. The total internal events core damage frequency (CDF) reported in the Quantification Notebook is 4.74E-05/yr [16].

Salem Unit 1 Internal Events Level 2 PRA Model

The Level 2 Model that is used for Salem Unit 1 was developed to calculate the LERF contribution as well as the other release categories evaluated in the model [17]. Table 4.2-1 summarizes the pertinent Salem Unit 1 results in terms of end-states. The total Large Early Release Frequency (LERF) in Table 4.2-1 was found to be 5.06E-6/yr with a total release frequency of 4.92E-05/yr. Note that the sum of the individual release categories is slightly higher than the reported CDF, but the individual release category frequencies are utilized here to provide the necessary delineation for the ILRT risk assessment with the corresponding EPRI/NEI class for each release category is listed in Table 4.2-1. The release categories are described after the table. The slight conservative treatment of total CDF used in this analysis will not significantly affect the overall results.

TABLE 4.2-1
RESULTS FOR DETAILED RELEASE CATEGORIES

RELEASE CATEGORY	EPRI/NEI	FREQUENCY	PERCENT
INTACT	1	8.94E-06	18%
LATE-BMMT-AFW	7	1.81E-10	<0.1%
LATE-BMMT-NOAFW	7	9.55E-07	2%
LATE-CHR-AFW	7	2.51E-08	<0.1%
LATE-CHR-NOAFW	7	3.42E-05	70%
LERF-ISLOCA	8	2.97E-08	<0.1%
LERF-CI	2	2.22E-07	0.5%
LERF-CFE	7	3.37E-08	<0.1%
LERF-SGTR-AFW	8	2.55E-06	5%
LERF-SGTR-NOAFW	8	1.98E-07	0.4%
LERF-ISGTR	8	2.03E-06	4%
Total	-	**4.92E-05**	-

Detailed Release Categories

The detailed release categories consider the initiating event, availability of auxiliary feedwater during the event, and the ultimate containment failure or bypass mode (if applicable). Each Level 2 sequence is mapped into one of these detailed release categories.

INTACT

This release category captures all of the INTACT sequences. Because the containment is essentially intact, sequence variations have a negligible impact on the release characteristics. Releases to the environment are via normal containment leakage.

LATE-BMMT-AFW

This release category captures sequences that result in basemat melt-through with feedwater available to the steam generators. Because basemat melt-through takes many days to erode the thick basemat at Salem, containment failure is assumed at 100 hours in the fission product release determination.

LATE-BMMT-NOAFW

This release category captures sequences that result in basemat melt-through without feedwater available to the steam generators. Because basemat melt-through takes many days to erode the thick basemat at Salem, containment failure is assumed at 100 hours in the fission product release determination.

LATE-CHR-AFW

This release category captures sequences that result in containment failure due to late overpressure with feedwater available to the steam generators.

LATE-CHR-NOAFW

This release category captures sequences that result in containment failure due to late overpressure without feedwater available to the steam generators.

LERF-ISLOCA

This release category captures sequences caused by an unisolated ISLOCA. Those sequences from LERF with ISLOCA initiating events contribute to this category.

LERF-CI

This release category captures sequences that result in containment isolation failure due to either valve failure or excessive pre-existing containment leakage. Containment failure due to pre-existing leakage is assumed at the start time of the scenario for the release calculations.

LERF-CFE

This release category captures sequences that result in early containment failure due to steam explosion, hydrogen burn, and/or direct containment heating at the time of vessel breach.

LERF-SGTR-AFW

This release category captures sequences caused by a steam generator tube rupture that have successful operation of auxiliary feedwater. With or without isolation of the ruptured steam generator, SGTR sequences with core damage provide a direct release path to the environment through the steam generator relief valves. Those sequences from LERF with SGTR initiating events and successful AFW contribute to this category.

LERF-SGTR-NOAFW

This release category captures sequences caused by a steam generator tube rupture that also have failed auxiliary feedwater. With or without isolation of the ruptured steam generator, SGTR sequences with core damage provide a direct release path to the environment through the steam generator relief valves. Those sequences from LERF with SGTR initiating events and failure of AFW contribute to this category.

LERF-ISGTR

This release category captures sequences that result in either a pressure-induced or thermally-induced steam generator tube rupture that bypasses containment.

Population Dose Calculations

The population dose is calculated using Salem SAMA Evaluation for Units 1 & 2 to be submitted to the NRC in 2009 [9] and the most recent Level 2 Analysis results [17]. The dose rate and the release category frequencies from the SAMA Evaluation are used to calculate the dose for each release category. After adjusting for projected 2015 population data (compared to the 2040 projected data used in the SAMA analysis), the dose for each release category is obtained and then grouped as per Table 4.2-1 for the EPRI/NEI Classes. The groupings along with the release category frequencies are utilized to obtain a weighted average dose per EPRI/NEI Class for use in the analysis.

Table 4.2-2 summarizes the information from the SAMA analysis that is used to determine the population dose for each release category. Note that this initial step is necessary since the latest Level 2 model results are used in the assessment which is based on a slightly lower total CDF and associated release frequency of 4.92E-05/yr (compared to 4.95E-05/yr used at the time of the SAMA analysis).

TABLE 4.2-2
CALCULATION OF SGS POPULATION DOSE RISK AT 50 MILES [9]

RELEASE CATEGORIES	POPULATION DOSE RISK [PERSON-REM/YR, MEAN]	RELEASE CATEGORY FREQUENCIES (PER YEAR)	POPULATION DOSE (PERSON-REM) (1)
INTACT	1.51E-01	9.22E-06	1.64E+04
LATE-BMMT-AFW	1.51E-05	1.81E-10	8.33E+04
LATE-BMMT-NOAFW	2.28E-02	9.89E-07	2.31E+04
LATE-CHR-AFW	6.35E-02	2.52E-08	2.52E+06
LATE-CHR-NOAFW	4.28E+01	3.42E-05	1.25E+06
LERF-ISLOCA	6.15E-01	2.97E-08	2.07E+07
LERF-CI	2.32E+00	2.23E-07	1.04E+07
LERF-CFE	3.71E-01	3.40E-08	1.09E+07
LERF-SGTR-AFW	2.32E+01	2.55E-06	9.10E+06
LERF-SGTR-NOAFW	7.82E-01	1.98E-07	3.95E+06
LERF-ISGTR	7.94E+00	2.03E-06	3.91E+06
Total	**7.82E+01**	**4.95E-05**	**1.58E+06**

(1) Obtained by dividing the population dose risk shown in the second column by the release category frequency in the third column of this table.

Population Estimate Methodology

The 50-mile population is predicted based on calculating the 10 year growth factor for radial intervals using 1990 and 2000 census data from the SECPOP2000 code. These growth factors calculated for 16 different directions are then applied to project out to year 2015 as shown in Table 4.2-3. The use of the 2015 estimate based on the more recent SECPOP2000 data is judged to be sufficient to perform this assessment.

TABLE 4.2-3
CALCULATION OF SGS POPULATION AT 50 MILES IN 2015

SECTOR (GROWTH FACTORS)	0-10 MILES(1)	10-20 MILES (1.16)(2)	20-30 MILES (1.09)(2)	30-40 MILES (1.01)(2)	40-50 MILES (1.04)(2)	50-MILE TOTAL
N	1124	155876	157614	157237	180342	**652,192**
NNE	9774	16890	130417	939315	1173404	**2,269,800**
NE	2835	10542	75484	405573	469580	**964,015**
ENE	2470	5450	36460	77764	40243	**162,386**
E	1146	41639	83655	21636	45822	**193,898**
ESE	1028	11186	17259	9685	25521	**64,680**
SE	0	89	641	0	43002	**43,732**
SSE	79	68	1416	1370	6917	**9,850**
S	785	17700	68311	26908	16739	**130,443**
SSW	798	20586	12420	9576	15237	**58,617**
SW	2905	4515	5941	6146	11230	**30,738**
WSW	3531	4515	3943	10859	32101	**54,948**
W	10701	6076	4542	54151	187489	**262,958**
WNW	4756	26817	28278	29628	24998	**114,478**
NW	3914	122265	32778	27538	46488	**232,982**
NNW	27668	150874	87312	74016	58368	**398,239**
Total	**73,514**	**595,090**	**746,471**	**1,851,400**	**2,377,481**	**5,643,956**

(1) Growth factors for 0-3 miles are 1; however, for 3-4 miles, 4-5 miles and 5-10 miles the growth factors as 1.19, 1.38 and 1.17, respectively. These have been applied to the corresponding 2000 SECPOP data and summed to yield the 0-10 miles results.

(2) Growth factors applied to the corresponding 2000 SECPOP data yield the values presented in the column.

The person-rem results in Table 4.2-2 are for the year 2040 as they were developed as a part of the license renewal effort. From the Salem SAMA Analysis [9] it is estimated that the 2040 population is 6.42E+06. A "Population Dose Factor" is calculated in order to convert the 2040 dose in person-rem to 2015 dose as shown below.

Total Salem 2015 Population 50 miles in Table 4.2-3 = 5.64E+06

Total Salem 2040 Population 50 miles in Salem SAMA [9] = 6.42E+06

Population Dose Factor = 5.64E+06 / 6.42E+06 = 0.88

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. This is considered adequate since the conclusions from this analysis will not be substantially affected by the actual dose values that are used.

Table 4.2-4 shows the results of applying the 2015 population dose factor to the population dose results in Table 4.2-2 at 50 miles. Furthermore, the most recent frequencies from the Level 2 analysis [17] are included to yield the best estimate 2015 applicable dose risk in person-rem/yr.

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

RELEASE CATEGORY	2040 POPULATION DOSE (PERSON-REM)	2015 POPULATION DOSE (PERSON-REM)(1)	RELEASE CATEGORY FREQUENCIES (PER YEAR)	POPULATION DOSE RISK [PERSON-REM/YR, MEAN] (2)
INTACT	1.64E+04	1.44E+04	8.94E-06	1.29E-01
LATE-BMMT-AFW	8.33E+04	7.33E+04	1.81E-10	1.33E-05
LATE-BMMT-NOAFW	2.31E+04	2.03E+04	9.55E-07	1.94E-02
LATE-CHR-AFW	2.52E+06	2.22E+06	2.51E-08	5.57E-02
LATE-CHR-NOAFW	1.25E+06	1.10E+06	3.42E-05	3.76E+01
LERF-ISLOCA	2.07E+07	1.82E+07	2.97E-08	5.41E-01
LERF-CI	1.04E+07	9.15E+06	2.22E-07	2.03E+00
LERF-CFE	1.09E+07	9.59E+06	3.37E-08	3.23E-01
LERF-SGTR-AFW	9.10E+06	8.01E+06	2.55E-06	2.04E+01
LERF-SGTR-NOAFW	3.95E+06	3.48E+06	1.98E-07	6.88E-01
LERF-ISGTR	3.91E+06	3.44E+06	2.03E-06	6.98E+00
Total	**1.58E+06**	**1.39E+06**	**4.92E-05**	**6.88E+01**

(1) The 2015 population dose per event is estimated from the calculated 2040 population dose multiplied by the 0.88 population factor as described above.

(2) Obtained by multiplying the 2015 population dose in the third column by the most recent Level 2 frequencies in the fourth column [17].

4.3 IMPACT OF EXTENSION ON DETECTION OF COMPONENT FAILURES THAT LEAD TO LEAKAGE (SMALL AND LARGE)

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

The probability of the EPRI Class 3a failures may be determined, consistent with the latest EPRI guidance [22], 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 [22], a non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e. 0.5/(217+1) = 0.0023).

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

The supplemental information states:

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

in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage."

The application of this additional guidance to the analysis for Salem Unit 1 (as detailed in Section 5) means that the Class 2, Class 7-LERF and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the Salem Unit 1 Level 2 PRA analysis. Additionally, the LERF-CFE category assigned to release category Class 7 from Table 4.2-1 (referred to as Class 7-LERF in this report) is excluded since it also already contributes to LERF.

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

Salem Unit 1 Past ILRT Results

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

Based on completion of two successful ILRTs at Salem Unit 1, the current ILRT interval is once per ten years. Note that the probability of a pre-existing leakage due to extending the ILRT interval is based on the industry wide historical results as discussed in the NEI Guidance document, and the only portion of Salem Unit 1 specific information utilized is the fact that the current ILRT interval is once per ten years.

NEI Interim Guidance

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

1. Quantify the baseline (nominal three year ILRT interval) frequency per reactor year for the EPRI accident categories of interest. Note that EPRI categories 4, 5, and 6 are not affected by changes in the ILRT test frequency.
2. Determine the containment leakage rates for EPRI categories 1 and 3 where category 3 is subdivided into categories 3a and 3b for "small" and "large" isolation failures, respectively.
3. Develop the baseline population dose (person-rem) for the applicable EPRI categories.
4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in Step (3) by the associated frequency calculated in Step (1).
5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however, the non-detection probability of leakage detectable only by ILRT does increase.
6. Determine the population dose rate for the new surveillance intervals of interest.
7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.
8. Evaluate the risk impact in terms of LERF.
9. Evaluate the change in conditional containment failure probability.

The first seven steps of the methodology calculate the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF for

Salem Unit 1, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The ninth and final step of the interim methodology calculates the change in containment failure probability, referred to as the conditional containment failure probability, CCFP. The NRC has previously accepted similar calculations [7] for the change in CCFP as the basis for showing that the proposed change is consistent with the defense in depth philosophy. As such, this last step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174.

4.4 IMPACT OF EXTENSION ON DETECTION OF STEEL LINER CORROSION THAT LEADS TO LEAKAGE

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [19]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. Salem Unit 1 has a similar containment type.

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 used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to the Salem Unit 1 containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.
- For consistency with the Calvert Cliffs analysis, the estimated historical flaw probability is limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional data that is available since the time of the Calvert analysis has not been factored into this analysis to maintain consistency with other submittals and since it is judged to have a minimal impact on the results (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 are determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the ILRT target pressure of 37 psig. For Salem Unit 1, the containment failure probabilities are less than these values at the ILRT test pressure of 47 psig [24]. Conservative probabilities of 1% for the cylinder and dome, and 0.1% for the basemat is 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.)
- For Salem Unit 1 there is approximately a 34% (percentage of area visually inaccessible) inspection detection failure likelihood as discussed in the Salem Unit 2's ILRT extension report [23] and the fact that the Salem Unit 2's containment is very similar to Unit 1's. In addition, an innate 5% failure of the process itself at detecting a failure is assumed giving a total of 39%. 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 34% and 44%, 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="5">2</td><td rowspan="5">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></td><td><u>Failure Rate</u></td><td><u>Year</u></td><td><u>Failure Rate</u></td></tr>
<tr><td>1</td><td>2.1E-3</td><td>1</td><td>5.0E-4</td></tr>
<tr><td>avg 5-10</td><td>5.2E-3</td><td>avg 5-10</td><td>1.3E-3</td></tr>
<tr><td>15</td><td>1.4E-2</td><td>15</td><td>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 [19]).</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)
1.02% (1 to 10 years)
2.35% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, 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 while also accouting for the unique arrangement of the Salem containment. (Note that the 34% number is conservatively applied to the cylinder and wall and 100% is applied to the basemat even though the 34% number derived for the Unit 2 submittal accounted for the cylinder, dome, and basemat in its detemrination. Assigning the 34% to the cylinder and dome is therefore conservative, but will not significantly impact the results of the analysis,)	**39%** 34% 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. A 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.

TABLE 4.4-1
STEEL LINER CORROSION BASE CASE

STEP	DESCRIPTION	CONTAINMENT CYLINDER AND DOME	CONTAINMENT BASEMAT
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4 * 5)	**0.0028% (at 3 years)** =0.71% * 1% * 39% **0.0158% (at 10 years)** =4.06% * 1% * 39% **0.0367% (at 15 years)** =9.40% * 1% * 39%	**0.00018% (at 3 years)** =0.18% * 0.1% * 100% **0.00102% (at 10 years)** =1.02% * 0.1% * 100% **0.00235% (at 15 years)** =2.35% * 0.1% * 100%

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

At 3 years : 0.0028% + 0.00018% = 0.00298%
At 10 years: 0.0158% + 0.00102% = 0.01682%
At 15 years: 0.0367% + 0.00235% = 0.03905%

5.0 RESULTS

The application of the approach based on NEI Interim Guidance [3, 21], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6, 7, 20, 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5.0-1 lists these accident classes.

The analysis performed examined Salem Unit 1-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the break down of the severe accidents contributing to risk were considered in the following manner:

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

TABLE 5.0-1
ACCIDENT CLASSES

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

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

Step 1 Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5.0-1.

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

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

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

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

5.1 STEP 1 – QUANTIFY THE BASE-LINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

This step involves the review of the Salem Unit 1 Level 2 release category frequency results. As described in Section 4.2, the release categories were assigned to the EPRI/NEI classes as seen in Table 4.2-1. This application combined with the Salem Unit 1 2015 dose (person-rem) results determined from Table 4.2-4 forms the basis for estimating the population dose for Salem.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. These events are represented by the Class 3 sequences in EPRI TR-104285. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The initial set of containment release class frequencies as shown in Table 4.2-4 are developed consistent with the definitions in Table 5.0-1 and have been reproduced below in Table 5.1-1.

TABLE 5.1-1
CALCULATION OF SGS POPULATION DOSE RISK AT 50 MILES

RELEASE CATEGORY	EPRI/NEI CLASS	2015 POPULATION DOSE (PERSON-REM)	RELEASE CATEGORY FREQUENCIES (PER YEAR)	POPULATION DOSE RISK [PERSON-REM/YR, MEAN]
INTACT	1	1.44E+04	8.94E-06	1.29E-01
LATE-BMMT-AFW	7	7.33E+04	1.81E-10	1.33E-05
LATE-BMMT-NOAFW	7	2.03E+04	9.55E-07	1.94E-02
LATE-CHR-AFW	7	2.22E+06	2.51E-08	5.57E-02
LATE-CHR-NOAFW	7	1.10E+06	3.42E-05	3.76E+01
LERF-ISLOCA	8	1.82E+07	2.97E-08	5.41E-01
LERF-CI	2	9.15E+06	2.22E-07	2.03E+00
LERF-CFE	7	9.59E+06	3.37E-08	3.23E-01
LERF-SGTR-AFW	8	8.01E+06	2.55E-06	2.04E+01
LERF-SGTR-NOAFW	8	3.48E+06	1.98E-07	6.88E-01
LERF-ISGTR	8	3.44E+06	2.03E-06	6.98E+00
Total	-	**1.39E+06**	**4.92E-05**	**6.88E+01**

In order to group the release categories by EPRI/NEI classes the summation of the dose risks for a given class is divided by the total frequency for that class. This yields frequency weighted average dose for the class. The doses or weighted average doses (when applicable) are shown in Table 5.1-2.

TABLE 5.1-2
CONTAINMENT RELEASE TYPE ASSIGNMENT

EPRI TR-104285 CONTAINMENT RELEASE	
SCENARIO TYPE	**DOSE (PERSON-REM)**
1	1.44E+04
2	9.15E+06
7	1.07E+06
7-LERF	9.59E+06
8	5.96E+06

The frequencies for the severe accident classes defined in Table 5.0-1 are developed for Salem Unit 1 based on the assignments shown above in Table 5.1-2, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments are made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4.

<u>Class 1 Sequences</u>

This group represents the frequency when the containment remains intact (modeled as Technical Specification Leakage). The frequency per year for these sequences is 8.43E-06/yr and is determined by subtracting all containment failure end states including the EPRI/NEI Class 3a and 3b frequency calculated below, from the total CDF. For this analysis, the associated maximum containment leakage for this group is 1La, consistent with an intact containment evaluation.

Class 2 Sequences

This group consists of large containment isolation failures. For Salem Unit 1 this frequency is 2.22E-07/yr.

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 on those cases that are already LERF scenarios (i.e. the Class 2, Class 7-LERF and Class 8 contributions).

Class_3a = 0.0092 * (CDF - Class 2 - Class 7LERF - Class 8)
= 0.0092 * (4.92E-05 – 2.22E-07 -3.37E-08 – 4.81E-06)
= 4.06E-07/yr

Class_3b = 0.0023 * (CDF - Class 2 - Class 7LERF - Class 8)
= 0.0023 * (4.92E-05 – 2.22E-07 -3.37E-08 – 4.81E-06)
= 1.01E-07/yr

For this analysis, the associated containment leakage for Class 3a is 10La and for Class 3b is 100La. These assignments are consistent with the latest EPRI Guidance.

Class 4 Sequences

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 the analysis.

Class 5 Sequences

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.

Class 6 Sequences

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 NEI Interim Guidance, however, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences

This group represents containment failure induced by severe accident phenomena. The failure frequency for non-LERF and LERF sequeces is shown below in Table 5.1-3. The total release frequency and total dose are then used to determine a weighted average person-rem for use as the representative EPRI Class 7 dose. Note that the total frequency and dose associated with this EPRI class does not change as part of the ILRT extension request.

TABLE 5.1-3
ACCIDENT CLASS 7 FAILURE FREQUENCIES AND POPULATION DOSES (SALEM UNIT 1 BASE CASE LEVEL 2 MODEL)

ACCIDENT CLASS	RELEASE FREQUENCY/YR	POPULATION DOSE (50 MILES) PERSON-REM (1)	POPULATION DOSE RISK (50 MILES) (PERSON-REM/YR) (2)
7 non-LERF	3.52E-05	1.07E+06	3.77E+01
7 LERF	3.37E-08	9.59E+06	3.23E-01
Class 7 Total	3.52E-05	1.08E+06 (3)	3.80E+01

(1) Population dose values obtained from Table 4.2-4

(2) Obtained by multiplying the release frequency value from the second column of this table by the population dose value from the third column of this table.

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

Class 8 Sequences

This group represents sequences when containment bypass occurs. For Salem Unit 1 this frequency is 4.81E-06/yr.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definition of Accident Classes defined in EPRI-TR-104285 shown in Table 5.1-4.

TABLE 5.1-4
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (SALEM BASE CASE)

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (PER RX-YR)
1	No Containment Failure	8.43E-06
2	Large Isolation Failures (Failure to Close)	2.22E-07
3a	Small Isolation Failures (liner breach)	4.06E-07
3b	Large Isolation Failures (liner breach)	1.01E-07

TABLE 5.1-4
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (SALEM BASE CASE)

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (PER RX-YR)
4	Small Isolation Failures (Failure to seal –Type B)	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A
6	Other Isolation Failures (e.g. dependent failures)	N/A
7	Failures Induced by Phenomena	3.52E-05
7-LERF	Failures Induced by Phenomena (LERF)	3.37E-08
8	Bypass (Interfacing System LOCA)	4.81E-06
CDF	**All CET End states (including very low and no release)**	**4.92E-05**

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

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by the Salem SAMA Analysis to be submitted [9] and the Level 2 Analysis [17] as described in Section 4.2, and summarized in Table 4.2-4. The results of applying these releases to the EPRI/NEI containment failure classification are as follows:

Class 1 = 1.44E+04 person-rem (at 1.0La)

Class 2 = 9.15E+06 person-rem

Class 3a = 1.44E+04 person-rem x 10La = 1.44E+05 person-rem

Class 3b = 1.44E+04 person-rem x 100La = 1.44E+06 person-rem

Class 4 = Not analyzed

Class 5 = Not analyzed

Class 6 = Not analyzed

Class 7 = 1.07E+06 person-rem

Class 7-LERF = 9.59E+06 person-rem

Class 8 = 5.96E+06 person-rem

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [2] containment failure classifications, and consistent with the NEI guidance [3, 21] are provided in Table 5.2-1.

TABLE 5.2-1
SALEM POPULATION DOSE ESTIMATES
FOR POPULATION WITHIN 50 MILES

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (50 MILES)
1	No Containment Failure (1 La)	1.44E+04
2	Large Isolation Failures (Failure to Close)	9.15E+06
3a	Small Isolation Failures (liner breach)	1.44E+05
3b	Large Isolation Failures (liner breach)	1.44E+06
4	Small Isolation Failures (Failure to seal –Type B)	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA
6	Other Isolation Failures (e.g. dependent failures)	NA
7	Failures Induced by Phenomena	1.07E+06
7-LERF	Failures Induced by Phenomena (LERF)	9.59E+06
8	Bypass (Interfacing System LOCA)	5.96E+06

The above dose estimates when multiplied by the frequency results presented in Table 5.1-4 yield the Salem Unit 1 baseline mean consequence measures for each accident class. These results are presented in Table 5.2-2. The calculated dose for Salem Unit 1 compares favorably with other locations given the relative population densities surrounding each location.

TABLE 5.2-2
SALEM ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (50 MILES)	NEI METHODOLOGY		NEI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION PERSON-REM/YR[1]
			FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	
1	No Containment Failure [2]	1.44E+04	8.43E-06	1.22E-01	8.43E-06	1.22E-01	-1.88E-05
2	Large Isolation Failures (Failure to Close)	9.15E+06	2.22E-07	2.03E+00	2.22E-07	2.03E+00	--
3a	Small Isolation Failures (liner breach)	1.44E+05	4.06E-07	5.86E-02	4.06E-07	5.86E-02	--
3b	Large Isolation Failures (liner breach)	1.44E+06	1.01E-07	1.46E-01	1.03E-07	1.48E-01	1.88E-03
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g. dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena	1.07E+06	3.52E-05	3.77E+01	3.52E-05	3.77E+01	--

TABLE 5.2-2
SALEM ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (50 MILES)	NEI METHODOLOGY		NEI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION PERSON-REM/YR[1]
			FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	
7-LERF	Failures Induced by Phenomena (LERF)	9.59E+06	3.37E-08	3.23E-01	3.37E-08	3.23E-01	
8	Bypass (ISLOCA)	5.96E+06	4.81E-06	2.86E+01	4.81E-06	2.86E+01	--
CDF	All CET end states		4.92E-05	6.90E+01	4.92E-05	6.90E+01	1.86E-03

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

5.3 STEP 3 – EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10-TO-15 YEARS

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

Risk Impact Due to 10-year Test Interval

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

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 5.3-2.

TABLE 5.3-1
SALEM UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (50 MILES)	NEI METHODOLOGY		NEI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION PERSON-REM/YR[1]
			FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	
1	No Containment Failure [2]	1.44E+04	7.25E-06	1.22E-01	7.24E-06	1.05E-01	-1.07E-04
2	Large Isolation Failures (Failure to Close)	9.15E+06	2.22E-07	2.03E+00	2.22E-07	2.03E+00	--
3a	Small Isolation Failures (liner breach)	1.44E+05	1.35E-06	1.95E-01	1.35E-06	1.95E-01	--
3b	Large Isolation Failures (liner breach)	1.44E+06	3.38E-07	4.88E-01	3.45E-07	4.98E-01	1.07E-02
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A
6	Other Isolation Failures (e.g. dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena	1.07E+06	3.52E-05	3.77E+01	3.52E-05	3.77E+01	--

TABLE 5.3-1
SALEM UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS;
CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (50 MILES)	NEI METHODOLOGY		NEI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION PERSON-REM/YR[1]
			FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	
7-LERF	Failures Induced by Phenomena (LERF)	9.59E+06	3.37E-08	3.23E-01	3.37E-08	3.23E-01	
8	Bypass (ISLOCA)	5.96E+06	4.81E-06	2.86E+01	4.81E-06	2.86E+01	--
CDF	**All CET end states**		**4.92E-05**	**6.94E+01**	**4.92E-05**	**6.94E+01**	**1.06E-02**

(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.

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

TABLE 5.3-2
SALEM UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (50 MILES)	NEI METHODOLOGY		NEI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION PERSON-REM/YR[1]
			FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	
1	No Containment Failure [2]	1.44E+04	6.40E-06	9.24E-02	6.39E-06	9.22E-02	-2.48E-04
2	Large Isolation Failures (Failure to Close)	9.15E+06	2.22E-07	2.03E+00	2.22E-07	2.03E+00	--
3a	Small Isolation Failures (liner breach)	1.44E+05	2.03E-06	2.93E-01	2.03E-06	2.93E-01	--
3b	Large Isolation Failures (liner breach)	1.44E+06	5.07E-07	7.32E-01	5.25E-07	7.57E-01	2.48E-02
4	Small Isolation Failures (Failure to seal –Type B)	NA	N/A	N/A	N/A	N/A	N/A
5	Small Isolation Failures (Failure to seal—Type C)	NA	N/A	N/A	N/A	N/A	N/A

TABLE 5.3-2
SALEM UNIT 1 ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	PERSON-REM (50 MILES)	NEI METHODOLOGY		NEI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION PERSON-REM/YR[1]
			FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	FREQUENCY (PER RX-YR)	PERSON-REM/YR (50 MILES)	
6	Other Isolation Failures (e.g. dependent failures)	NA	N/A	N/A	N/A	N/A	N/A
7	Failures Induced by Phenomena	1.07E+06	3.52E-05	3.77E+01	3.52E-05	3.77E+01	--
7-LERF	Failures Induced by Phenomena (LERF)	9.59E+06	3.37E-08	3.23E-01	3.37E-08	3.23E-01	
8	Bypass (ISLOCA)	5.96E+06	4.81E-06	2.86E+01	4.81E-06	2.86E+01	--
CDF	**All CET end states**		**4.92E-05**	**6.98E+01**	**4.92E-05**	**6.98E+01**	**2.46E-02**

(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.

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

5.4 STEP 4 – DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr, and small changes in LERF as below 1E-6/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Salem Unit 1, 100% of the frequency of Class 3b sequences can be used as a conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the NEI guidance methodology). Based on the original 3/10 year test interval assessment from Table 5.2-2, the Class 3b frequency is 1.03E-07/yr, which includes the corrosion effect of the containment liner. Based on a ten-year test interval from Table 5.3-1, the Class 3b frequency is 3.45E-07/yr; and, based on a fifteen-year test interval from Table 5.3-2, it is 5.25E-07/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years (including corrosion effects) is 4.22E-07/yr. Similarly, the increase due to increasing the interval from 10 to 15 years (including corrosion effects) is 1.79E-07/yr. As can be seen, even with the conservatisms included in the evaluation (per the NEI methodology), the estimated change in LERF is below the threshold criteria for a small change in risk when comparing the 15 year results to the current 10-year requirement, and even to the original 3-in-10 year requirement.

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

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

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

CCFP 3 IN 10 YRS	CCFP 1 IN 10 YRS	CCFP 1 IN 15 YRS	$\Delta CCFP_{15\text{-}3}$	$\Delta CCFP_{15\text{-}10}$
82.03%	82.53%	82.89%	0.86%	0.36%

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

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

5.6 SUMMARY OF INTERNAL EVENTS RESULTS

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

TABLE 5.6-1
SALEM UNIT 1 ILRT CASES:
BASE, 3 TO 10, AND 3 TO 15 YR EXTENSIONS
(INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF/YR	PER-REM/YR	CDF/YR	PER-REM/YR	CDF/YR	PER-REM/YR
1	1.44E+04	8.43E-06	1.22E-01	7.24E-06	1.05E-01	6.39E-06	9.22E-02
2	9.15E+06	2.22E-07	2.03E+00	2.22E-07	2.03E+00	2.22E-07	2.03E+00
3a	1.44E+05	4.06E-07	5.86E-02	1.35E-06	1.95E-01	2.03E-06	2.93E-01
3b	1.44E+06	1.03E-07	1.48E-01	3.45E-07	4.98E-01	5.25E-07	7.57E-01

TABLE 5.6-1
SALEM UNIT 1 ILRT CASES:
BASE, 3 TO 10, AND 3 TO 15 YR EXTENSIONS
(INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF/YR	PER-REM/YR	CDF/YR	PER-REM/YR	CDF/YR	PER-REM/YR
7	1.07E+06	3.52E-05	3.77E+01	3.52E-05	3.77E+01	3.52E-05	3.77E+01
7-LERF	9.59E+06	3.37E-08	3.23E-01	3.37E-08	3.23E-01	3.37E-08	3.23E-01
8	5.96E+06	4.81E-06	2.86E+01	4.81E-06	2.86E+01	4.81E-06	2.86E+01
Total		4.92E-05	6.90E+01	4.92E-05	6.94E+01	4.92E-05	6.98E+01
ILRT Dose Rate from 3a and 3b		2.07E-01		6.93E-01		1.05E+00	
Delta Total Dose Rate[1]	From 3 yr	---		4.69E-01		8.14E-01	
	From 10 yr	---		---		3.44E-01	
3b Frequency (LERF)		1.03E-07		3.45E-07		5.25E-07	
Delta LERF	From 3 yr	---		2.43E-07		4.22E-07	
	From 10 yr	---		---		1.79E-07	
CCFP %		82.03%		82.53%		82.89%	
Delta CCFP %	From 3 yr	---		0.49%		0.86%	
	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 because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g. the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the IRLT frequency.

5.7 EXTERNAL EVENTS CONTRIBUTION

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

The method chosen to account for external events contributions is similar to that used in the SAMA analysis [9] to use a multiplier on the internal events results based on the IPEEE methodology [18]. The contributions of the external events from the original IPEEE analysis are summarized in Table 5.7-1.

TABLE 5.7-1
ORIGINAL IPEEE CONTRIBUTOR SUMMARY

EXTERNAL EVENT INITIATOR GROUP	CDF
Seismic	9.5E-06 per yr
Internal Fire	2.3E-05 per yr
High Winds	Not Applicable (progressive screening method used)
External Floods	3E-07 per yr(1)
Transportation and Nearby Facility Accidents	6.7E-08 per yr [18]
Detritus	5.2E-07 per yr to 9.2E-07 per yr [18]
Chemical Release	Not Applicable (progressive screening method used)
Total (for initiators with CDF available)	**3.4E-05 per yr**

(1) A progressive screening method used and an overall CDF was not calculated, but three potential water ingress paths were estimated to contribute CDFs of about 1E-07 each.

From Table 5.7-1 the external events multiplier could be calculated as external events CDF divided by the internal events CDF as (3.40E-05/yr) / (4.92E-05/yr) = 0.69. Based on other considerations described in the SAMA analysis, the end result was to use a multiplier of 1.0. That multiplier will also be conservatively used for this 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 5.6-1 of the PRA analysis portion of the submittal as 1.03E-07/yr, 3.45E-07/yr, and 5.25E-07/yr, respectively. Therefore, the change in the LERF risk measure due to extending the ILRT from 3-per-10 years to 1-

per-15 years, including both internal and external hazard risk, is estimated as shown in Table 5.7-2.

TABLE 5.7-2
SGS1 3B (LERF) AS A FUNCTION OF ILRT FREQUENCY
FOR INTERNAL AND EXTERNAL EVENTS
(INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

	3B FREQUENCY (3-PER-10 YR ILRT)	3B FREQUENCY (1-PER-10 YEAR ILRT)	3B FREQUENCY (1-PER-15 YEAR ILRT)	LERF INCREASE(1)
Internal Events Contribution	1.03E-07	3.45E-07	5.25E-07	4.22E-07
External Events Contribution (Internal Events CDF x 1.0)	1.03E-07	3.45E-07	5.25E-07	4.22E-07
Combined (Internal + External)	2.06E-07	6.91E-07	1.05E-06	8.44E-07

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

Thus, the total increase in LERF (measured from the original 3 per 10 years required to the proposed 1 per 15 years performance of the ILRT) due to the combined internal and external events contribution is estimated as 8.44E-07/yr.

NRC Regulatory Guide 1.174 [4], "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant. As discussed in Section 2 of this PRA analysis, the risk acceptance criteria of RG 1.174 is used here to assess the ILRT interval extension.

The 8.44E-07/yr increase in LERF due to the combined internal and external events from extending the Salem ILRT frequency from 3-per-10 years to 1-per-15 years falls within Region II between 1E-7 to 1E-6 per reactor year ("Small Change" in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the "Small Change" range, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr. Similar bounding assumptions regarding the external event contributions that were made above are used for the total LERF estimate.

From Table 4.2-1, the LERF due to postulated internal event accidents is 5.06E-06/yr. However, much of this total, i.e. 2.78E-06/yr, is due to ISLOCA and SGTR initiators and would therefore not be applicable to external events initiators. Because of this, the base LERF due to external events is reduced by this amount to take this into account.

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

Internal Events LERF	5.06E-06/yr
External Events LERF	2.29E-06/yr
Internal Events LERF due to ILRT (at 15 years)(1)	4.22E-07/yr
External Events LERF due to ILRT (at 15 years)(1)	4.22E-07/yr
Total	**8.19E-06/yr**

(1) Including age adjusted steel liner corrosion likelihood.

As can be seen, the estimated upper bound LERF for Salem is estimated at 8.19E-06/yr, which is less than the RG 1.174 requirement to demonstrate that the total LERF of internal events and external events is less than 1E-5/yr.

6.0 SENSITIVITIES

6.1 SENSITIVITY TO CORROSION IMPACT ASSUMPTIONS

The results in Tables 5.2-2, 5.3-1, and 5.3-2 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment. 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 39% to 44% and 34%. The results are presented in Table 6.1-1. In almost every case, the impact from including the corrosion effects is very minimal. Only the containment breach assumption has the potential to challenge the acceptance guidelines, but the base value was conservatively chosen to be close to the Calvert Cliff assumptions even though the likelihood of containment failure at the ILRT test pressure is much lower for Salem. Given this and the other conservative assumptions associated with the analysis, it is judged that the conclusions should not change.

TABLE 6.1-1
STEEL LINER CORROSION SENSITIVITY CASES

AGE (STEP 3 IN THE CORROSION ANALYSIS)	CONTAINMENT BREACH (STEP 4 IN THE CORROSION ANALYSIS)	VISUAL INSPECTION & NON-VISUAL FLAWS (STEP 5 IN THE CORROSION ANALYSIS)	INCREASE IN CLASS 3B FREQUENCY (LERF) FOR ILRT EXTENSION 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 (39% Cylinder-Dome, 100% Basemat)	4.22E-07	1.59E-08
Doubles every 2 yrs	Base	Base	4.42E-07	3.64E-08
Doubles every 10 yrs	Base	Base	4.19E-07	1.34E-08
Base	Base	44% Cylinder-Dome	4.24E-07	1.78E-08
Base	Base	34% Cylinder-Dome	4.20E-07	1.40E-08
Base	10% Cylinder-Dome, 1% Basemat	Base	5.65E-07	1.59E-07
Base	0.1% Cylinder-Dome, 0.01% Basemat	Base	4.07E-07	1.59E-09
LOWER BOUND				
Doubles every 10 yrs	1.0% Cylinder-Dome, 0.1% Basemat	34% Cylinder-Dome 100% Basemat	4.07E-07	1.18E-09
UPPER BOUND				
Doubles every 2 yrs	10% Cylinder-Dome, 1% Basemat	44% Cylinder-Dome 100% Basemat	8.13E-07	4.08E-07

6.2 EPRI EXPERT ELICITATION SENSITIVITY

An expert elicitation was performed to reduce excess conservatisms in the data associated with the probability of undetected leaks within containment [22]. Since the risk impact assessment of the extensions to the ILRT interval is sensitive to both the probability of the leakage as well as the magnitude, it was decided to perform the expert elicitation in a manner to solicit the probability of leakage as a function of leakage magnitude. In addition, the elicitation was performed for a range of failure modes which allowed experts to account for the range of failure mechanisms, the potential for undiscovered mechanisms, un-inspectable areas of the containment as well as the potential for detection by alternate means. The expert elicitation process has the advantage of considering the available data for small leakage events, which have occurred in the data, and extrapolate those events and probabilities of occurrence to the potential for large magnitude leakage events.

The basic difference in the application of the ILRT interval methodology using the expert elicitation is a change in the probability of pre-existing leakage in the containment. The base case methodology uses the Jeffrey's non-informative prior for the large leak size and the expert elicitation sensitivity study uses the results of the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the base case methodology (i.e. 10 La for small and 100 La for large) are used here. Table 6.2-1 illustrates the magnitudes and probabilities of a pre-existing leak in containment associated with the 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, and the input to expert elicitation as well as the results of the expert elicitation are available in the various appendices of the EPRI report [22].

TABLE 6.2-1
EPRI EXPERT ELICITATION RESULTS

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

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

The net effect is that the reduction in the multipliers shown above has the same impact on the calculated increases in the LERF values. The increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 4.36E-08/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is 1.82E-08/yr. As such, if the expert elicitation mean probabilities of occurrence are used instead of the non-informative prior estimates, the change in LERF for Salem Unit 1 is even below the threshold criteria for a "very small" change in risk when compared to the current 1-in-10 or original 3-in-10 year requirement. The results of this sensitivity study are judged to be more indicative of the actual risk associated with the ILRT extension than the results from the assessment as dictated by the NEI methodology values, and yet are still conservative given the assumption that all of the Class 3b contribution is considered to be LERF.

TABLE 6.2-2
SALEM UNIT 1 ILRT CASES:
BASE, 3 TO 10, AND 3 TO 15 YR EXTENSIONS
(BASED ON EPRI EXPERT ELICITATION LEAKAGE PROBABILITIES)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF/YR	PER-REM/YR	CDF/YR	PER-REM/YR	CDF/YR	PER-REM/YR
1	1.44E+04	8.76E-06	1.26E-01	8.33E-06	1.20E-01	8.03E-06	1.16E-01
2	9.15E+06	2.22E-07	2.03E+00	2.22E-07	2.03E+00	2.22E-07	2.03E+00
3a	1.44E+05	1.71E-07	2.47E-02	5.70E-07	8.23E-02	8.56E-07	1.24E-01
3b	1.44E+06	1.09E-08	1.57E-02	3.63E-08	5.24E-02	5.45E-08	7.86E-02
7	1.07E+06	3.52E-05	3.77E+01	3.52E-05	3.77E+01	3.52E-05	3.77E+01
7-LERF	9.59E+06	3.37E-08	3.23E-01	3.37E-08	3.23E-01	3.37E-08	3.23E-01
8	5.96E+06	4.81E-06	2.86E+01	4.81E-06	2.86E+01	4.81E-06	2.86E+01
Total		4.92E-05	6.88E+01	4.92E-05	6.89E+01	4.92E-05	6.90E+01
ILRT Dose Rate from 3a and 3b		4.04E-02		1.35E-01		2.02E-01	
Delta Total Dose Rate(1)	From 3 yr	---		8.81E-02		1.51E-01	
	From 10 yr	---		---		6.31E-02	
3b Frequency (LERF)		1.09E-08		3.63E-08		5.45E-08	
Delta LERF	From 3 yr	---		2.54E-08		4.36E-08	
	From 10 yr	---		---		1.82E-08	
CCFP %		81.85%		81.90%		81.93%	
Delta CCFP %	From 3 yr	---		0.052%		0.090%	
	From 10 yr	---		---		0.037%	

1. The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g. the EPRI Class 2, Class 7-LERF and Class 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the IRLT frequency.

7.0 CONCLUSIONS

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

- Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Small changes in risk are defined as increases in CDF below 10^{-5}/yr and increases in LERF below 10^{-6}/yr. Since the ILRT does not impact CDF for Salem, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years is estimated as 4.06E-07/yr (i.e. in the "small" change region using the acceptance guidelines of Reg. Guide 1.174) using the NEI guidance as written, and 4.36E-08/yr (i.e. in the "very small" change region) using the EPRI Expert Elicitation methodology. The increase in internal events LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years for the base case with corrosion included is 4.22E-07/yr which also falls in the "small" change region of the acceptance guidelines in Reg. Guide 1.174.
- The change in Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 7.91E-01 person-rem/yr using the NEI guidance, and drops to 1.51E-01 person-rem/yr using the EPRI Expert Elicitation methodology. Therefore, in either case, the risk impact when compared to other severe accident risks is negligible.
- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen year interval is about 0.83% using the NEI guidance, and drops to about 0.09% using the EPRI Expert Elicitation methodology. Although no official acceptance criteria exist for this risk metric, it is judged to be very small.
- To determine the potential impact from external events, an additional bounding assessment from the risk associated with external events utilizing the information from the Salem IPEEE was performed. As shown in Table 5.7-2, the total increase in LERF due to internal events and the bounding external events assessment is 8.44E-07/yr, which is in Region II of the Reg. Guide 1.174 acceptance guidelines.

- Finally, the same bounding analysis indicates that the total LERF from internal and external risks as shown in Table 5.7-3 is 8.19E-06/yr, which is less than the Reg. Guide 1.174 limit of 1E-05/yr given that the ΔLERF is in Region II.

Therefore, increasing the ILRT interval to 15 years is not considered to be significant since it represents a small change to the Salem Unit 1 risk profile.

Previous Assessments

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

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

The findings for Salem Unit 1 confirm these general findings on a plant specific basis considering the severe accidents evaluated for Salem Unit 1, the Salem containment failure modes, and the local population surrounding Salem.

8.0 REFERENCES

[1] Nuclear Energy Institute, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, NEI 94-01, July 1995.

[2] Electric Power Research Institute, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI TR-104285, August 1994.

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

[4] U.S. Nuclear Regulatory Commission, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, Revision 1, November 2002.

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

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[18] Salem Generating Station Individual Plant Examination of External Events, Public Service Electric and Gas Company, January 1996.

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

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Appendix A
PRA Technical Adequacy

A.1 Overview

A technical Probabilistic Risk Assessment (PRA) analysis is presented in this calculation to help support a one-time extension of the Salem Unit 1 containment Type A test integrated leak rate test (ILRT) interval from ten years to fifteen years.

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

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

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

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

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

A.2 Technical Adequacy of the PRA Model

The PRA model version used for the ILRT assessment is the Salem Unit 1 PRA, Revision 4.2. This model update was completed in March 2009. The Revision 4.2 update to the Salem Generating Station (SGS) PRA model is the most recent evaluation of the risk profile at Salem for internal event challenges. The SGS PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the SGS PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

The SGS PRA model is controlled in accordance with ER-AA-600-1015 "FPIE PRA Model Update" This procedure defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.

- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every three years.

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

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

A PRA updating requirements evaluation (URE- PRA model update tracking database) is created for all issues that are identified that could impact the PRA model. The URE database includes the identification of those plant changes that could impact the PRA model. A review of the current open items in the URE database for SGS identified no items with potential impact as the latest PRA model was just recently completed.

A.2.2 Applicability of Peer Review Findings and Observations

Several assessments of technical capability have been made for the SGS PRA model. These assessments are as follows and are further discussed below.

- The Salem Peer Certification occurred in December 2001 [2]. Subsequent updates to the SGS model have addressed all three of the "A" level findings and all 39 of the "B" level findings.
- During 2005 the SGS PRA model results were evaluated in the Pressurized Water Reactors Owners Group PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process [3]. SGS was initially identified as a candidate outlier, but that status was resolved with subsequent updates to the PRA model.
- A PRA Peer Review of the SGS PRA was performed at the end of 2008. The results of the PRA Peer Review indicated that a number of the supporting requirements (SRs) were "Not Met" or only met "Category I". As noted in the peer review report, most of these findings pertained to documentation issues and as such would not impact this

application, but the report did indicate that there were eight key findings. The potential impact of the eight key findings is discussed further in Section A.2.3. In summary, the results of the SGS PRA Peer Review support the quality of the SGS PRA and its use for this application.

The Salem Peer Certification occurred in December 2001 [2]. The model reviewed during the certification review was a draft version of Revision 3.0. The final update completed for Revision 3.0 addressed some of the comments identified during the certification process; i.e., all significance Level "A" comments and some Level "B" comments. Other "B" Facts and Observations (F&Os) were addressed in the subsequent Revision 3.1 model. With the exception of two 'B' F&Os, Revision 3.2a addressed the remaining F&Os. All A and B F&Os are now closed with version 4.2 (the version used as the basis for this assessment).

The SGS PRA was included in the Westinghouse Owners Group (WOG) MSPI cross-comparison [3]. Salem was listed as a candidate outlier in the High Pressure Injection and Auxiliary Feedwater Systems. The WOG cross-comparison was performed using model results from a draft revision of the SGS PRA (Revision 3.2). Since the cross-comparison was finished, the PRA model has been updated to address the outlier issues. In addition to the WOG cross-comparison, the NRC performed an analysis to determine outliers. Salem was determined to be an outlier for MD AFW Pump 12(22) and TD AFW Pump 13(23) due to low Birnbaum values. Because of the changes made to the PRA, the Birnbaum values for these pumps have increased and are now consistent with industry group values. The outlier status has been resolved with the NRC [4].

A.2.3 Consistency with Applicable PRA Standards

As indicated above, a formal peer review of the SGS PRA was performed in 2008 and the final peer review report issued in 2009 [5]. The Peer Review was performed against Addendum B of the PRA Standard [6], the criteria in RG-1.200, Revision 1 [1] including the NRC positions stated in Appendix A of RG-1.200, Revision 1 and further issue clarifications [7]. The results of the PRA Peer Review indicated that a number of the supporting requirements (SRs) were "Not Met" or only met "Category I". As noted in the peer review report, most of these findings pertained to documentation issues and as such would not impact this application, but the report did indicate that there were eight key findings. Descriptions of those findings are provided in Table A.2-1 along with an assessment of the impact for this application.

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

TABLE A.2-1
KEY FINDINGS FROM THE SALEM PRA 2008 PEER REVIEW

ELEMENT	DESCRIPTION	IMPACT ON APPLICATION
AS	The first was that the ISLOCA sequence with no piping failure is assumed to be terminated with operator isolation of the suction path using the pump suction isolation MOVs. However, isolation cannot be accomplished until primary pressure is reduced. The potential for flooding of adjacent areas by water lost through the RHR pump seals and/or RHR heat exchangers prior to isolation does not appear to have been evaluated. The significance of this is that flooding of adjacent areas could impact additional equipment affecting the ability to achieve a safe, stable condition.	No impact. ISLOCA scenarios are by definition not subject to fission product retention in the containment that could be impacted by the change in the ILRT interval. In addition, the current model does not credit isolation of and recovery from RHR suction ISLOCAs that result in leakage into the RHR pump area.
DA	The second issue involved data and specifically component availability. Component availability depends on an accurate count of maintenance unavailability (DA-C11). Maintenance and testing unavailability were identified in the model. However, no specific surveillance tests were discussed in the Data Analysis Notebook. MSPI/Maintenance Rule sources were identified. The specific surveillances or plant maintenance contributing to the unavailability of plant components and the process for counting these durations should be documented in a data procedure.	Non-significant impact. The PRA data evaluation for Salem is based on MSPI and Maintenance Rule data, which is believed to be accurate. Any changes to plant-specific failure rates from a comparison of expected unavailability due totest procedures and maintenance with actual MSPI and Maintenance Rule data is expected to be non-significant.
IE	The third issue involves Initiating Events. For those initiators that are modeled using fault trees, such as loss of SW and loss of Capability Category, the initiator frequency is not based on reactor year. For example, under gate IE-TSW, basic event SWS-PIP-RP-TBHDR has a mission time of 8760 hours. Use reactor year which considers the actual plant availability is the expected metric when quantifying the initiator frequencies.	No impact. The current treatment is conservative. Addressing the issue would only lead to a small reduction in the calculated annualized CDF.

TABLE A.2-1
KEY FINDINGS FROM THE SALEM PRA 2008 PEER REVIEW

ELEMENT	DESCRIPTION	IMPACT ON APPLICATION
IE	The fourth issue involves Initiating Events. The Initiating Events Notebook describes the review of Salem Generating Station Experience and Trip Review. No mention is made of consideration of events that occurred at conditions other than at-power operation. Also, events resulting in controlled shutdown were excluded on the basis that they present only mild challenges rather than being determined to be not applicable to at-power operation. Failure to consider non-power events and controlled shutdown events could result in exclusion of valid initiating events.	Non-significant impact. The identification of the applicable initiating events for Salem did include a review of events other than at-power operations. Events occuring during shutdowns and non-power conditions which could have occurred at power were not excluded. The SGS PRA model includes a broad range of initiating events that are suffcient for this application.
IF	The fifth issue involves Internal Flooding. Flood scenarios were screened without development of flood rate, source, and operator actions. Detailed assessments were only provided for selected high-frequency floods. Improperly screening flood scenarios could lead to underestimating the risks associated with internal floods.	No impact. The requirements in IF-C2c and IF-C3 allow screening of flood areas. These requirements are in conflict with and, therefore, basically nullify the requirements of IF-C1, IF-C2, and IF-C2a. The treatment for Salem is consistent with what is noted in Section 4.5.7.1 of the standard that "Some degree of event and scenario screening is typically employed in analyzing risk from internal flooding, so that, although the high level and supporting requirements are written in a discrete manner, the requirements are not necessarily presented in sequential order of application and, in some cases, must be considered jointly, so that screening is performed appropriately."

TABLE A.2-1
KEY FINDINGS FROM THE SALEM PRA 2008 PEER REVIEW

ELEMENT	DESCRIPTION	IMPACT ON APPLICATION
AS	The sixth issue involves Accident Sequences. Specifically, the SBO success paths following offsite power recovery do not address recovery and operation of required safety systems after power recovery which is considered necessary to demonstrate that a safe, stable end state has been achieved. In addition, the combination of RCP Seal LOCA and offsite power recovery into a single top event treatment does not provide explicit treatment of the differences in recovery time and required mitigation response for different RCP seal leakage rates. More explicit development of the SBO event sequences will ensure that they represent a safe, stable end state and appropriately consider all required mitigation equipment.	Non-significant impact. The Salem offsite power recovery model considers the status of key equipment and also the potential for varying RCP seal leakage rates in determining the time available for offsite power recovery. The likelihood of LOOP, SBO, successful recovery of offsite power, then multiple equipment failures preventing long-term safe shutdown is very small. The current model provides an appropriate evaluation of risks associated with loss of offsite power events. This treatment provides a reasonable approximation of the SBO event sequence development that is sufficient for this application.
DA	The seventh was the omission of failure modes for the diesels due to the use of only MSPI data and not all the plant specific data. Plant-specific data is only collected for MSPI components. Documentation describing the process of collecting the number of failures, hours of operation, number of surveillance tests and planned maintenance activities on plant requirements could not be identified. Appendixes to the data notebook identify data collected, but the source was often not provided. Without this source of documentation future updates could be difficult.	Non-significant impact. The PRA data evaluation for Salem is based on MSPI and Maintenance Rule data. Data from plant programs is believed to be reliable. Any changes to plant-specific failure rates from a validation of other plant specific data with what is readily available from MSPI and Maintenance Rule data is expected to be non-significant.

TABLE A.2-1
KEY FINDINGS FROM THE SALEM PRA 2008 PEER REVIEW

ELEMENT	DESCRIPTION	IMPACT ON APPLICATION
DA	The eighth was the lack of defining system boundaries. A draft document was provided that documented how to establish component boundaries, how to establish failure probabilities, sources of generic data, etc. This procedure needs to be formalized. The notebook could be improved by providing direct references to actual failure numbers in EPIX or CDE numbers in the Data Notebook, Appendix A. Assumptions were noted in various sections of the Data Analysis Notebook. These need to be gathered into an assumptions section in the notebook. Sources of uncertainty were not discussed in the analysis.	No impact. This issues discussed in this key finding are documentation related issues.

A.2.4 Identification of Key Assumptions

The methodology employed in this risk assessment followed the NEI guidance updated with more recent data and utilized the same process that has been utilized in several similar relief requests (including an earlier request for Salem Unit 2). The analysis included the incorporation of several sensitivity studies and factored in the potential impacts from external events in a bounding fashion. Since the accepted process utilizes a bounding analysis approach which is mostly driven by that CDF contribution which does not already lead to LERF, there are no identified key assumptions or sources of uncertainty for this application (i.e. those which would change the conclusions from the risk assessment results presented here).

A.3 Summary

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

A.4 References

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

[2] Salem Generating Station Probabilistic Risk Assessment Peer Review Report, Final Report, December 2002.

[3] Westinghouse Owner's Group Mitigating Systems Performance Index Cross Comparison, WCAP-16464-NP, Revision 0, July 2005.

[4] NRC "MSPI Unresolved PRA Outliers and Generic PRA Issues Lists" as of January 20, 2006.

[5] RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Salem Generating Station, Units 1 & 2 Probabilistic Risk Assessment, Westinghouse Project PA-RMSC-0386, May 2009.

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

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