PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

JAN 1 7 2002
LRN-02-0001
LCR S02-01

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

Gentlemen:

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
ONE-TIME EXTENSION TO INCREASE THE INTERVAL OF THE INTEGRATED LEAK RATE TEST FROM TEN TO FIFTEEN YEARS
SALEM GENERATING STATION, UNIT 2
FACILITY OPERATING LICENSE DPR-75
DOCKET NO. 50-311

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to the Technical Specifications (TS) for the Salem Nuclear Generating Station Unit 2. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed amendment revises Technical Specifications 6.8.4.f, "Primary Containment Leakage Rate Testing Program." This will allow a one-time interval extension for the Salem Nuclear Generating Station Unit 2 - Type A, Integrated Leakage Rate Test (ILRT) for no more than five (5) years.

PSEG has evaluated in Attachment 1 the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined that the request involves no significant hazards considerations. In addition, there is no significant increase in the amounts or types of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed amendment satisfies the criteria of 10CFR51.22 (c)(9) for categorical exclusion from the requirement for an environmental assessment. The marked up Technical Specification page affected by the proposed change is provided in Attachment 2.

A plant-specific, risk-based evaluation (Attachment 3, Calculation S-C-ZZ-MEE-1613 Rev. 0, "Salem Generating Station 2 ILRT Extension") has been performed in support of the one-time extension to extend the Type A test from once in 10 years to once in 15 years. The guidance in NEI 94-01, Electric Power Research Institute (EPRI) Topical

A001

Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," and Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis," was used.

PSEG requests NRC approval of the proposed License Amendment by March 31, 2002 to be implemented within 30 days. The requested approval date and implementation period will allow sufficient time to reschedule the remaining outage activities to achieve optimum effectiveness of Refueling Outage 12 (2R12), scheduled to begin on April 05, 2002. The reason for this request is to save critical path time in 2R12 and move the ILRT to one of the three subsequent refueling outages where it can be performed off critical path. Removing the ILRT from 2R12 will reduce the critical path by approximately 44 hours with no significant effect on safety.

If you have any questions or require additional information, please contact Mr. Michael Mosier at (856) 339-5434.

Sincerely,

T. J. O'Connor
Vice President -
Nuclear Maintenance/Plant Support

Attachments (3)

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

Executed on 1/17/02

T. J. O'Connor
Vice President -
Nuclear Maintenance/Plant Support

C:

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS ONE-TIME EXTENSION TO INCREASE THE INTERVAL OF THE INTEGRATED LEAK RATE TEST FROM TEN TO FIFTEEN YEARS

1. DESCRIPTION

This letter is a request to amend Facility Operating License DPR-75 for the Salem Nuclear Generating Station, Unit 2 (SGS-2). The proposed change would revise Technical Specification 6.8.4.f, "Primary Containment Leakage Rate Testing Program" to permit a one-time extension to the maximum ten-year interval to fifteen years to perform the Type A test. The proposed change will provide an economic benefit by eliminating the Integrated Leak Rate Test (ILRT) from Refueling Outage 12 (2R12), reducing the critical path by approximately 44 hours with no significant impact on safety.

Approval of this proposed change is being requested by the end of March 2002 to support the scheduled implementation date of April 2002.

2. PROPOSED CHANGE

The proposed changes to the Technical Specifications (TS) are included in Attachment 2 of this submittal. In summary, it is requested that:

Section 6.8.4.f, "Primary Containment Leakage Rate Testing Program," be amended to permit a one-time extension to the maximum ten-year frequency to be increased to fifteen years to perform the ILRT. The proposed TS change is based on past successful Type A, B, and C tests, and American Society of Mechanical Engineers (ASME) Section XI inspections (reference 7.4) at SGS-2. The results for SGS-2 are shown in Table 1. Further justification is based on research documented in NUREG-1493 (reference 7.5) which generically shows that very few potential containment leakage paths fail to be identified by Type B and C tests. In fact, an analysis of 144 ILRT test results, including 23 failures, found that no failures were due to containment liner breach. The NUREG concluded that reducing the Type A (ILRT) testing frequency to once per twenty years would lead to an imperceptible increase in risk. A plant specific calculation provided in Attachment 3 demonstrates that the risk impact of the proposed change when compared to other severe accident risks is negligible. The purpose of this submittal is to request a one-time deferral of the Type A (ILRT) from March 24, 2002 to no later than March 23, 2007.

3. BACKGROUND

ILRTs have been required of operating nuclear power plants to ensure the public health and safety in the case of an accident that would release radioactivity to the containment. Conservative design and construction have led to very few ILRTs exceeding their required leakage. The NRC has extended the allowable ILRT test period from three times in ten years to once in ten years based on past successful tests. NUREG-1493 that supported the change to the ten-year interval also stated that test periods of up to twenty years would lead to an imperceptible increase in risk.

Section 3.8.1 of the Salem Updated Final Safety Analysis Report (UFSAR) describes the primary containment. 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 design objective of the containment structure is to contain all radioactive material which might be released from the core following a loss-of-coolant accident (LOCA). The structure serves as both a biological shield and a pressure container.

The basic structural elements considered in the design of the containment structure are the base slab, side walls, and dome acting as one structure under all possible loading conditions. The liner is anchored to the concrete shell by means of anchors so that it forms an integral part of the entire composite structure under all loadings. The reinforcing in the structure will have an elastic response to all loads with limited maximum strains to ensure the integrity of the steel liner. The lower portions of the cylindrical liner are insulated to avoid buckling of the liner due to restricted radial growth when subjected to a rise in temperature.

The primary containment 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 bottom liner plate, in the annulus area between the circular crane wall and the outer cylindrical wall, is covered with a minimum of 2 feet of concrete, and the area within the crane wall is covered with 5 feet of concrete. The top of these concrete slabs is the floor of the containment. The base mat is directly supported on lean concrete fill.

The satisfactory results from previous integrated leakage rate tests at SGS-2, as well as continued satisfactory results of local leak rate tests, and containment inspections, support deferral of the 2R12 test. The reactor containment building will continue to be inspected under the requirements of ASME Section XI Subsections IWE and IWL. The existing Type B and C containment penetration-testing program will continue to be performed in accordance with previous regulatory approvals.

PSEG has performed three operational ILRT tests. All tests passed the as-found acceptance criteria of 1.0 L_a, where L_a is the maximum allowable accident leakage rate. The results are shown in Table 1.

Structural degradation of containment is a gradual process that occurs due to the effects of pressure, temperature, radiation, chemical, or other such effects. Such effects would be identified and corrected when the containment structure is periodically tested and inspected to verify structural integrity under ASME Section XI Subsections IWE and IWL. The most recent visual inspections performed were during refueling outage 2R11, in October 2000. These surveillances provide a high degree of assurance that any degradation of the containment structure will be detected and corrected before it can produce a containment leakage path. The tests and inspections conducted to date have not identified degradation that threatens the integrity of the SGS-2 containment.

The NRC has approved similar changes in Amendment No. 197 for Crystal River Unit 3. Also, in Amendment No. 206 for Entergy Nuclear Operations, Inc.'s Indian Point Nuclear Generating Unit 3 the NRC approved a one-time increase 10 to 15 years for the ILRT interval.

4. TECHNICAL ANALYSIS

The purpose of this analysis is to demonstrate the risk of extending the Type A (ILRT) interval from the current 10 years required by 10 CFR 50, Appendix J (reference 7.7) at Salem Generating Station (SGS-2) to 15 years is negligible. The baseline risks in this analysis are defined in terms of population dose (person-rem) per reactor year, large early release frequency (LERF) and conditional containment failure probability (CCFP). Consequently, the impact of type A extension is evaluated against the person-rem, LERF and CCFP. Our methodology differs from the IPE level 2 results in that we excluded release sequences that are not impacted by the Type A test in the LERF calculation. Also, credit for containment visual inspection was utilized.

This calculation evaluates the risk associated with various ILRT intervals as follows. The focus is on the changes in risk from the current 10 years to the proposed 15 years.

3 years	interval based on the original requirements of 3 tests per 10 years
10 years	current test interval required for SGS-2
15 years	interval extension approved for Indian Point 3 (IP3), Crystal River 3 (CR3) and proposed for SGS-2

4.1 Methodology

The evaluation for SGS-2 is consistent with similar assessments performed for CR3 (reference 7.8), and IP3 (reference 7.9) with enhancement outlined in the EPRI Interim Guidance (reference 7.10). The IP3 and CR3 submittals have been approved by the NRC (references 7.8 & 7.11). These assessments utilize the guidelines set forth in NEI 94-01 (reference 7.2), the methodology used in EPRI TR-104285 (reference 7.6) and NUREG-1493 (reference 7.5) and the regulatory guidance on the use of Probabilistic Safety Assessment (PSA) findings in support of a licensee request to change a plant's licensing basis, RG 1.174 (reference 7.3). The calculation applies the SGS-2 Individual Plant Examination (IPE) plant damage states and the Level 3 PRA person-rem estimates in order to estimate the changes in risk due to increasing the ILRT test interval. This information is obtained from the SGS-2 IPE (reference 7.12) and a Level 3 PRA study (reference 7.13) performed by SCIENTECH for SGS-2.

In addition to the methodology approved by the NRC and suggested by EPRI, two improvements have been implemented in this evaluation. These improvements are specifically used for LERF calculation. The first improvement addresses the LERF contributors. It was determined that only a portion of the core damage sequences will be impacted by the Type A test in their relationship to the large early release. For instance, the containment bypass coupled with core damage will not be impacted by the Type A test. In other words, the portion of the Core Damage Frequency (CDF) that contributes to LERF should not be double counted[1]. Another example is the late core melt (after the general evacuation). Since the potential release occurs after population evacuation, it is, by definition, not a LERF contributor. In this evaluation, the core damage sequences are examined to determine if the Type A test will impact the outcome of the large early release.

The second improvement is incorporation of the visual inspection (IWL and IWE) into the evaluation. To be conservative, this analysis credits detections of large liner failures only. For small failures, no credit was given to the visual inspection. With no evidence of large failures and thus no history of detections, it is conservatively estimated that the visual inspection will have at a 50% probability of finding a large liner failure.

The analysis steps performed are listed below:

- Calculate the Level 3 release category population dose frequencies.
- Map the Level 3 release categories into the 8 release classes defined by the EPRI report.
- Calculate the Type A leakage estimate to define the analysis baseline.
- Calculate the Type A leakage estimate to address the current inspection frequency.
- Calculate the Type A leakage estimates to address extension of the Type A test interval.
- Calculate increase in risk due to extending Type A inspection intervals.
- Calculate the change in LERF due to extending the Type A testing interval.
- Calculate the change in conditional containment failure probability due to extending the Type A testing interval.

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[1] This point is actually noted in the CR3 and IP2 applications. For instance, the CR3 evaluation assumption number 7 states, "The containment releases for Classes 2, 6, 7, and 8 are not impacted by the ILRT Type A test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A."

4.2 Assumptions/Bases

- The maximum containment leakage for Class 1 sequences is estimated using the level 3 PRA results and is defined as 1 L_a unit for this analysis.
- The maximum containment leakage for Class 3a sequences is 10 times the class 1 sequences based on the previously approved methodology (references 7.9 & 7.11).
- The maximum containment leakage for Class 3b sequences is 35 times the class 1 sequences based on the previously approved methodology (references 7.9 & 7.11).
- Containment leakage due to Classes 4 and 5 are considered negligible based on the previously approved methodology (references 7.9 & 7.11).
- The containment releases are not impacted with time.
- The containment releases for Classes 2, 6, 7, and 8 are not impacted by the ILRT Type A test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A.
- Because Class 8 sequences are containment bypass sequences, potential releases are directly to the environment. Therefore, the containment structure will not impact the release magnitude.

4.3 Calculation

The inputs for this calculation come from the information documented in the SGS-2 IPE (reference 7.12) and a Level 3 PSA study performed by SCIENTECH for SGS-2 (reference 7.13). The Level 3 study used the MACCS2 computer code to develop person-rem dose results. The study also used site-specific inputs for meteorological and population data. Other inputs to this calculation include ILRT test data from NUREG-1493 (reference 7.5), EPRI Interim Guidance (reference 7.10) and EPRI report TR-104285 (reference 7.6), and are referenced in the body of the calculation.

The current SGS-2 PRA is a non-safety-related tool and is intended to provide "best-estimate" results that can be used as input when making risk-informed decisions. The SGS-2 IPE (reference 7.12) is an earlier version of the PRA submitted in response to Generic Letter 88-20. Neither the PRA nor the IPE is considered as design basis information.

4.4 Risk Impact

The change in ILRT test frequency from once every ten years to once every fifteen years increases the total integrated plant risk by only 0.31 %. Also, the change in Type A test frequency from the original every three years to once every fifteen years increases the risk only 0.75%. Therefore, the risk impact when compared to other severe accident risks is negligible.

Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7} per year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the ILRT test frequency from the current once every 10 years to once every 15 years is 5.28×10^{-8} per year. Therefore, risk significance is very small.

Reg. Guide 1.174 recommends the use of risk analysis techniques to ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 0.45% for the proposed change and 1.08% for the cumulative change of going from a test frequency of every three years to once every fifteen years. These changes are small and demonstrate that the defense-in-depth philosophy is maintained. The results are summarized in Table 2.

Table 1
Salem Nuclear Generating Station Unit 2
ILRT Results

Test Date	Total Time Method Leakage Rate (%/day)	Test Duration (Hours)	TS 3.3.3 Acceptance Criteria
May 23, 1983	0.048* 0.054**	35	0.75 L_a
November 27, 1986	0.032* 0.043**	24	0.75 L_a
March 24, 1992	0.03196 (as found) 0.0224 (as left)	24	0.75 L_a

* Measured leakage

** Leakage with 95% upper confidence

Table 2

Summary of Risk Impact on Extending Type A ILRT Test Frequency

	Risk Impact for 3-year interval (baseline)	**Risk Impact for 10-year interval (current requirement)**	**Risk Impact for 15-year interval (proposed)**
Total Integrated Risk (Person-Rem/yr)	37.54	37.71	37.82
Type A Testing Risk (Person-Rem/yr)	0.0764	0.254	0.382
% Total Risk (Type A / Total)	0.20%	0.68%	1.01%
Type A LERF (Class 3b) (per year)	6.33E-08	1.06E-07	1.58E-07
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			0.11
% Increase from current (Δ Risk / Total Risk)			0.31%
Δ LERF from current (per year)			5.28E-08
Δ CCFP from current			0.45%
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			0.28
% Increase from baseline (Δ Risk / Total Risk)			0.75%
Δ LERF from baseline (per year)			9.50E-08
Δ CCFP from baseline			1.08%

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration Determination

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10CFR50.92, "Issuance of amendment," as discussed below:

1. **Does not involve a significant increase in the probability or consequences of an accident previously analyzed?**

Response: No.

The proposed revision to Section 6.8.4.f adds a one-time extension to the current interval for containment integrated leak rate test (ILRT). The current test interval of 10 years, based upon past performance, would be extended on a one-time basis to 15 years from the last ILRT. The proposed extension to ILRT testing cannot increase the probability of an accident previously evaluated since the containment ILRT testing extension is not a modification to plant systems, nor a change to plant operation that could initiate an accident. The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," found that very few potential containment leakage paths fail to be identified by Type B and C tests. The NUREG concluded that reducing the ILRT testing frequency to once per twenty years would lead to an imperceptible increase in risk. Containment performance monitoring is performed in accordance with the Maintenance Rule (10CFR50.65) and inspections required by American Society of Mechanical Engineers (ASME) code are performed in order to identify indications of containment degradation that could affect leak tightness. Type B and C testing required by the technical specifications (TS) will identify any containment opening, such as valves, that would otherwise be detected by the ILRT. Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. It also recommends the use of risk analysis techniques to ensure and show that the proposed change is consistent with the defense-in-depth philosophy. The increase in large early release fraction (LERF) resulting from a change in the ILRT test frequency from the current once every 10 years to once every 15 years is 5.28E-8 per year. The change in conditional containment failure probability (CCFP) was estimated to be 0.45% for the proposed change. These factors show that an ILRT test extension will not represent a significant increase in the consequences of an accident.

Therefore, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed.

2. **Does not create the possibility of a new or different kind of accident from any accident previously analyzed?**

Response: No

The proposed revision to Section 6.8.4.f adds a one-time exception to the current interval for the ILRT. The current test interval of 10 years, based upon past performance, would be extended on a one-time basis to 15 years from the last Type A test. Primary containment is designed to contain energy and fission products during and after an event. The Individual Plant Examination (IPE) identifies events that lead to containment failure. Revision to the ILRT test interval does not change this list of events. There are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating a new or different kind of accident. Therefore, this proposed amendment does not create the possibility of a new or different kind of accident from any previously analyzed.

3. **Does not involve a significant reduction in the margin of safety?**

Response: No

The proposed revision to Section 6.8.4.f adds a one-time extension to the current interval for the ILRT. The current test interval of 10 years, based upon past performance, would be extended on a one-time basis to 15 years from the last ILRT. The proposed extension to ILRT testing interval will not significantly reduce the margin of safety. The NUREG-1493 generic study of the effects of extending containment leakage testing found that a 20-year exception in ILRT leakage testing resulted in an imperceptible increase in risk to the public. NUREG-1493 found that the containment leakage rate contributes a very small amount to the individual risk, and that the decrease in Type A testing frequency would have a minimal affect on this risk since most potential leakage paths are detected by Type C testing. Type B and Type C testing will continue to be performed at a frequency currently required by the Technical Specifications (TS). The containment inspections being performed in accordance with ASME, Section XI, and Maintenance Rule (10CFR50.65) provide a high degree of assurance that the containment will not degrade in a manner that is only detectable by Type A testing.

Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. It also recommends the use of risk analysis techniques to ensure and show that the proposed change is consistent with the defense-in-depth philosophy. The increase in large early release fraction (LERF) resulting from a change in the ILRT test frequency from the current once every 10 years to once every 15 years is 5.28E-8 per year. The change in conditional containment failure probability (CCFP) was estimated to be 0.45% for the proposed change.

Therefore, these changes do not involve a significant reduction in margin of safety.

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

5.2 Applicable Regulatory Requirements/Criteria

5.2.1 Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995 (RG 1.163).

5.2.2 Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" July 1998.

5.2.3 NUREG-1493, "Performance-Based Containment Leak-Test Program," Final Report, September 1995 (NUREG-1493).

5.2.4 Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors".

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

6. ENVIRONMENTAL IMPACT EVALUATION

PSEG 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. REFERENCES

7.1 Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995 (RG 1.163).

7.2 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).

7.3 Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," July 1998.

7.4 American Society of Mechanical Engineers (ASME) Section XI Subsections IWE and IWL, (Reactor Building Containment Inspections).

7.5 NUREG-1493, "Performance-Based Containment Leak-Test Program," Final Report, September 1995 (NUREG-1493).

7.6 EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals" (EPRI TR-104285).

7.7 Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors".

7.8 Florida Power, 3F0601-06, "Crystal River – Unit 3 – License Amendment Request #267, Revision 2, Supplemental Risk-Informed Information in Support of License Amendment Request #26," June 20, 2001.

7.9 Entergy, IPN-01-007, Indian Point 3 Nuclear Power Plant, "Supplemental Information Regarding Proposed Change to Section 6.14 of the Administrative Section of the Technical Specification," January 18, 2001.

7.10 Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Developed for NEI by EPRI, Jack Haugh, EPRI Area Manager, Nuclear Safety & Analysis John M. Gisclon, EPRI Consultant William Parkinson and Ken Canavan, Data Systems and Solutions November 2001

7.11 United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No.3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC NO. MB0178), April 17, 2001.

7.12 Salem Units 1 and 2 Probabilistic Risk Assessment Individual Plant Examination Submittal, Revision 0, July 1993.

7.13 Scientech 17268-001, "Salem MACCS2 Model and Level 3 Application," 12/18/2001.

7.14 NRC letter to PSEG issuing License Amendment 188, dated February 27, 1998 to implement the requirements of 10 CFR 50, Appendix J, Option B for performance-based primary containment leakage rate testing program (LCR S97-27).

7.15 1998 Edition of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," of Section XI, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

7.16 NEI Memo to the USNRC, "One-time extensions of containment integrated leak rate test interval – additional information," November 30, 2001.

SALEM GENERATING STATION-UNIT 2
FACILITY OPERATING LICENSE DPR-75
DOCKET NO. 50-311
REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

TECHNICAL SPECIFICATION PAGE WITH PROPOSED CHANGE

The following Technical Specification for Facility Operating License DPR-75 are affected by this change request:

Technical Specification	Page
6.8.4.f	6-19

(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. Backup Method for Determining Subcooling Margin

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. Postaccident Sampling

A program* which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel

(ii) Procedures for sampling and analysis,

(iii) Provisions for maintenance of sampling and analysis equipment.

6.8.4.f. Primary Containment Leakage Rate Testing Program

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. 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 following testing in accordance with this program, the leakage rate

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*It is acceptable if the licensee maintains details of the program in plant operation manuals.

Insert A

, as modified by the following exception:

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after March 24, 1992 shall be performed no later than March 24, 2007.

Attachment 3
LRN-02-0001

LCR S02-01

SALEM GENERATING STATION-UNIT 2
FACILITY OPERATING LICENSE
DOCKET NO. 50-311

Calculation S-C-ZZ-MEE-1613 Rev. 0
Salem Generating Station 2 ILRT Extension

FORM 1
Page 2 of 2
(Page 1 contains the instructions)

CALC NO.: S-C-ZZ-MEE-1613 **REVISION: 0**			**CALCULATION COVER SHEET**			**Page 0 of 25**	
CALC. TITLE:	PRA Analysis of Salem Generation Station ILRT EXtension						
# SHTS (CALC):	26	**# ATT / # SHTS:**		**# IDV/50.59 SHTS:**	26	**# TOTAL SHTS:**	52

CHECK ONE:

x☐ FINAL ☐ INTERIM (Proposed Plant Change) ☐ FINAL (Future Confirmation Req'd) ☐ VOID

SALEM OR HOPE CREEK: S☐ Q – LIST ☐ IMPORTANT TO SAFETY x☐ NON-SAFETY RELATED

HOPE CREEK ONLY: ☐Q ☐Qs ☐Qsh ☐F ☐R

☐ STATION PROCEDURES IMPACTED, IF SO CONTACT SYSTEM MANAGER

☐ CDs INCORPORATED (IF ANY): ______________________

DESCRIPTION OF CALCULATION REVISION (IF APPL.):

N/A

PURPOSE:

The purpose of this calculation is to estimate the risk associated with extending the Type A Integrated Leak Rate Test (ILRT) interval from the current 10 years required by 10 CFR 50, Appendix J [1] at Salem Generating Station (SGS) to 15 years.

CONCLUSIONS: The change in Type A test frequency from once every ten years to once every fifteen years increases the risk impact on the total integrated plant risk by only 0.31%. Also, the change in Type A test frequency from the original every three years to once every fifteen years increases the risk only 0.75%. Therefore, the risk impact when compared to other severe accident risks is negligible. Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from the current once every 10 years to once every 15 years is 5.28E-8/yr. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from 10 to 15 years is therefore considered non-risk significant. The LERF increase is 9.50E-8 for the cumulative change of going from a test frequency of three every ten years to once every fifteen years, which is still non-risk significant. R.G. 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 0.45% for the proposed change and 1.08% for the cumulative change of going from a test frequency of three every ten years to once every fifteen years. These changes are small and that the defense-in-depth philosophy is maintained.

	Printed Name / Signature	**Date**
ORIGINATOR/COMPANY NAME:	Wei He	01/04/02
PEER REVIEWER/COMPANY NAME:	N/A	N/A
VERIFIER/COMPANY NAME:	T. L. Gann	1/04/02
PSEG SUPERVISOR APPROVAL:		1/4/02

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1.0 PURPOSE/SCOPE

The purpose of this calculation is to demonstrate that the risk is negligible by extending the Type A Integrated Leak Rate Test (ILRT) interval from the current 10 years required by 10 CFR 50, Appendix J [1] at Salem Generating Station (SGS) to 15 years. The risk in this analysis is defined in terms of population dose (person-rem) per reactor year, large early release (LERF) and conditional containment failure probability (CCFP). Consequently, the impact of Type A extension is evaluated against the person-rem, LERF and CCFP. The results will be used to support a plant license amendment (PLA).

This calculation evaluates the risk associated with various ILRT intervals as follows. The focus is the risk changes from the current 10 years to the proposed 15 years.

- 3 years - interval based on the original requirements of 3 tests per 10 years
- 10 years – current test interval required for SGS
- 15 years – interval extension approved for Indian Point 3, Crystall River, etc. and proposed for SGS

2.0 REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors".
2. Florida Power, 3F0601-06, "Crystal River – Unit 3 – License Amendment Request #267, Revision 2, Supplemental Risk-Informed Information in Support of License Amendment Request #267," June 20, 2001.
3. Entergy, IPN-01-007, Indian Point 3 Nuclear Power Plant, "Supplemental Information Regarding Proposed Change to Section 6.14 of the Administrative Section of the Technical Specification", January 18, 2001.
4. United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No.3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC NO. MBO178), April 17, 2001.
5. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J", July 26, 1995, Revision 0
6. EPRI TR-104285, "Risk Assessment of Revised Containment Leak Rate Testing Intervals" August 1994.
7. NUREG-1493, "Performance-Based Containment Leak-Test Program", July 1995.
8. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" July 1998.
9. Salem Units 1 and 2 Probabilistic Risk Assessment Individual Plant Examination Submittal, Revision 0, July, 1993.
10. Scientech 17087-001, "Salem MACCS2 Model and Level 3 Application," 12/2001.
11. EPRI Interim Guidance, Rev. 4, November, 2001.

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12. 1998 Edition of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," of Section XI, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).
13. NEI Memo to the USNRC, 'One-time extensions of containment integrated leak rate test interval – additional information.' November 30, 2001.

3.0 METHODOLOGY

The evaluation for SGS is consistent with similar assessments performed for Crystal River 3 (CR3) [2], Indian Point 3 (IP3) [3] with enhancement outlined in the EPRI Interim Guidance [11]. The IP3 [4] and CR3 submittals were recently approved by the NRC. These assessments utilize the guidelines set forth in NEI 94-01 [5], the methodology used in EPRI TR-104285 [6] and NUREG-1493 [7] and the regulatory guidance on the use of Probabilistic Safety Assessment (PSA) findings in support of a licensee request to a plant's licensing basis, RG 1.174 [8]. The calculation applies the SGS IPE release categories and the Level 3 PRA person-rem estimates to estimate the changes in risk due to increasing the ILRT test interval. This information is obtained from the SGS IPE [9] and a Level 3 PRA study [10] performed by SCIENTECH for SGS.

In addition to references mentioned above, improvements suggested in the references [11] and [13] are implemented in this evaluation. The previous 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[1], 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.

Frequency 3b=(3b Failure probability)*(CDF minus CDF with independent LERF minus CDF that cannot cause LERF)

Salem has in place additional programs to provide for defense in depth relative to containment failure, including IWE/IWL and maintenance inspections of the containment. People familiar with the containment inspection program suggested that the visual inspection ought to detect concrete and liner failures. To be on the conservative side, this analysis credits detection of large liner failures. (More technical discussions are carried out in LERF section.)

It should be noted that the calculations are carried out using the MS Excel Spreadsheet. The round offs are carried through. Hand calculation of a single equation may yield a slightly different value.

The basic analysis steps are listed below:

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[1] This point is noted in CR3 and IP2 application. The CR3 evaluation assumption number 7 states that "The containment releases for Classes 2, 6, 7, and 8 are not impacted by the ILRT Type A test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A."

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1. Calculate the Level 3 release category population doses.
2. Map the Level 3 release categories into the 8 release classes defined by the EPRI report.
3. Calculate the Type A leakage estimate to define the analysis baseline.
4. Calculate the Type A leakage to address the current inspection frequency.
5. Calculate the Type A leakage estimates to address extension of the Type A test interval.
6. Calculate the change in population dose due to extending Type A inspection intervals.
7. Calculate the change in LERF due to extending Type A inspection intervals.
8. Calculate the change in CCFP due to extending Type A inspection intervals.

4.0 ASSUMPTIONS/BASES

1. The maximum containment leakage for Class 1 sequences is estimated using the level 3 PRA results and is defined as 1 La unit for this analysis.
2. The maximum containment leakage for Class 3a sequences is 10 times the class 1 sequences based on the previously approved methodology [2,3].
3. The maximum containment leakage for Class 3b sequences is 35 times the class 1 sequences based on the previously approved methodology [2,3].
4. Containment leakage due to Classes 4 and 5 are considered negligible based on the previously approved methodology [2,3].
5. The containment releases are not impacted with time.
6. The containment releases for Classes 2, 6, 7, and 8 are not impacted by the ILRT Type A test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A.
7. Because Class 8 sequences are containment bypass sequences, potential releases are directly to the environment. Therefore, the containment structure will not impact the release magnitude.
8. The IPE results will be used for following two reasons. First, the level 1 model is under revision and to incorporate comments from WOG certification. Second, the level 2 PRA for IPE is extensive and has enough information to support the calculation. The IPE CDF value is 6.27E-5/year.

5.0 CALCULATION

The current SGS PRA is a non-safety-related tool and is intended to provide "best-estimate" results which can be used as input when making risk-informed decisions. The SGS IPE [9] is an earlier version of the PRA submitted to

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NRC in response to Generic Letter 88-20. The current PRA is under revision. Neither the PRA nor the IPE is considered as design basis information. Other inputs to this calculation include ILRT test data from NUREG-1493 [7], EPRI Interim Guideline [11] and the EPRI report [6], and are referenced in the body of the calculation.

Step 1 - Calculate the Level 3 release category population dose frequencies.

Table 1 provides release categories with descriptions and person-rem for each category. The release category and its description come from the information documented in the SGS IPE [9]. The person-rem comes from a Salem specific Level 3 PSA study [11]. The Level 3 study used the MACCS2 computer code to develop person-rem dose results. The study also used site-specific inputs for meteorological and population data.

Table 1

Level 3 PRA Person-Rem Estimates By Release Category [10]

Release Category ID	Description	Person-Rem
NOMLTNR[2]	No core melt, or release	3.34E03
NOSTFL	No containment structural failure, no release	3.34E03
LGLTDI	Large, late containment failure with direct release	1.22E06
LGLTSS	Large, late containment failure with subsoil release	1.96E05
SMLATE	Small, controlled late containment failure	4.00E05
LGERDI	Large, early containment failure with direct release	3.46E06
LGERSS	Large, early containment failure with subsoil release	1.20E06
SGTRVO	SGTR with unlimited release (stuck open SG valve)	4.23E06
VSMUS	V-sequence with small, unscrubbed release	0[3]
VSMSC	V-sequence with small, scrubbed (fire spray) release	0[4]
BMMT	Very late containment basemat melt through	0[5]

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[2] No population dose was calculated for this category. The population dose was conservatively assumed to be the same as the NOSTFL category.

[3] Very low frequency and the release has been conservatively added to the large release category VLGUS.

[4] Very low frequency and the release has been conservatively added to the large release category VLGCS.

[5] The basemat melt through category is traditionally not assessed for population dose. In addition, the frequency of

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VLGUS	V-sequence with large, unscrubbed release	3.75E06
VLGCS	V-sequence with large, scrubbed release	2.50E06
SMERLY	Small, controlled early containment failure	4.84E06
SGTRVC	SGTR with small release (SG valves close)	0[6]

For comparison, the IP3 analysis [3] assumed that the doses were a function of the DBA LOCA leakage (La) using the following factors listed in Table 2.

Table 2
Indian Point Assumed Dose Factors [3][7]

Class	Dose Factor
1	1 La
2	35 La
3a	10 La
3b	35 La
4	0
5	0
6	35 La
7	100 La

To be able to derive the population dose, the frequency of the release category is needed and is summarized in table 3 below. The information is in SGS IPE, Table 4.10-6.

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the category is low (9.06E-8/year).

[6] Very low frequency and the release has been conservatively added to the large release category SGTRVO.

[7] In cases that the level 3 does not provide the release dose for a category, this table will be used.

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Table 3
Release Category Frequencies Based on IPE Results

Release Category	Frequency
NOMLTNR	1.42E-5
NOSTFL	1.52E-5
LGLTDI	1.48E-5
LGLTSS	8.33E-6
SMLATE	4.41E-6
LGERDI	3.70E-6
LGERSS	1.13E-6
SGTRVO	2.00E-7
VLGUS	1.01E-7
VLGSC	1.01E-7
SMERLY	6.13E-8
VSMUS	1.80E-7
VSMSC	1.80E-7
BMMT	9.06E-8
SGTRVC	4.56E-8
Total	**6.27E-5**

By combining the above tables 1 and 3, the population dose can be derived for each release category.

Step 2: Map IPE release categories into the 8 release classes defined by the EPRI Report [6]

EPRI Report TR-104285 defines eight (8) release classes as follows:

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Table 4 EPRI Containment Failure Classifications

Class 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 La, under Appendix J for that plant. The allowable leakage rates (La), are typically 0.1 weight percent of containment volume per day for PWRs and 0.5 weight percent per day for BWRs (all measured at Pac, calculated peak containment pressure related to the design basis accident). Changes to leak rate testing frequencies do not affect this classification.
Class 2	Containment isolation failures (as reported in the IPEs) include those accidents in which the pre-existing leakage is due to failure to isolate the containment. These include those that are dependent on the core damage accident in progress (e.g., initiated by common cause failure or support system failure of power) and random failures to close a containment path. Changes in Appendix J testing requirements do not impact these accidents.
Class 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. This accident class is applicable to sequences involving ILRTs (Type A tests) and potential failures not detectable by LLRTs.
Class 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.
Class 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.
Class 6	Containment isolation failures include those leak paths not identified by the LLRTs. The type of penetration failures considered under this class includes those covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program. This failure to isolate is not typically identified in LLRT. Changes in Appendix J LLRT test intervals do not impact this class of accidents.
Class 7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
Class 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 typically impact these accidents, particularly for PWRs.

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Table 5 presents the SGS release category mapping for these eight accident classes. Person-Rem per year is the product of the frequency and the Person-Rem.

Table 5 EPRI Classification of SGS Release Category Data

Category	Frequency/yr	Person-Rem	Person-Rem/Yr	EPRI Class
NOMLTNR	1.42E-5	3.34E03	4.75E-02	1
NOSTFL	1.52E-5	3.34E03	5.08E-02	1
LGLTDI	1.48E-5	1.22E06	1.81E01	7
LGLTSS	8.33E-6	1.96E05	1.63E00	7
SMLATE	4.41E-6	4.00E05	1.76E00	7
LGERDI.I[8]	1.88E-6	3.46E06	6.51E00	2
LGERDI.F	1.81E-6	3.46E06	6.27E00	7
LGERSS	1.13E-6	1.20E06	1.35E00	7
SGTRVO	2.00E-7	4.23E06	8.46E-01	8
VSMUS	1.80E-7	0	0	8
VSMSC	1.80E-7	0	0	8
BMMT	9.06E-8	N/A	N/A	N/A
VLGUS	1.01E-7	3.75E06	3.79E-1	8
VLGSC	1.01E-7	2.50E06	2.53E-1	8
SMERLY	6.13E-8	4.84E06	2.97E-1	7
SGTRVC	4.56E-8	0	0	8

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

As displayed in Table 5, the SGS IPE did not identify release categories specifically associated with EPRI Classes 3, 4, 5, or 6. Therefore, each of these classes are evaluated for applicability to SGS.

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[8] The release category LGERDI contains EPRI categories 2 and 7. Thus, it is separated into LGERDI.I and LGERDI.F with respective frequencies (section 4.10.1 of the IPE contains more detailed information).

Class 3:

Containment failures in this class are due to leaks such as liner breaches which would be detected by performing a Type A ILRT or visual inspection (IWE) as required by ASME code.

For this estimation, the question on containment isolation was modified consistent with the previously approved methodology [2,3], to include the probability of a liner breach (due to excessive leakage) at the time of core damage. Using this methodology, Class 3 is divided into two classes. These are Class 3a (small liner breach) and Class 3b (large liner breach).

To calculate the probability that a liner leak will be large (Class 3b), use was made of the data presented in NUREG-1493 [7] and new data presented by the EPRI Interim Guidance[11]. One data set found in NUREG-1493 reviewed 144 ILRTs and the EPRI Interim Guidance reviewed additional 38 ILRTs. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (La). Since 21 La does not constitute a large release, no large releases have occurred based on the 144 ILRTs reported in NUREG-1493. One failure was found in 38 ILRTs and was discussed in EPRI Interim Guidance and this failure was not considered large.

Because no class 3b failure has occurred in 182 ILRT tests, the EPRI Interim Guidance suggested that the Jeffery's non-informative prior distribution would be appropriate for the class 3b distribution (The rational for using the Jeffery's non-informative prior distribution was discussed in reference 11.)

Failure probability = (# of failures (0) + ½)/(Number of tests (182) + 1)

The number of large failures is zero and the probability is 0.5/183 = 0.0027

Therefore the frequency of release due to Class 3b failures is calculated as:

$$FREQ_{class3b} = PROB_{class3b} \times CDF = 0.0027 \times 6.27E\text{-}05/yr = 1.69E\text{-}07/yr$$

It should be noted that the above calculation is very conservative. Not all core damage progression will contribute to class 3b failure[9]. This point will be further discussed in LERF calculation section.

To calculate the probability that a liner leak will be small (Class 3a), use was made of the data presented in NUREG-1493 [7] and the EPRI Interim Guidance. The NUREG-1493 states that 144 ILRTs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of 1.0La. However, of these 23 'failures,' only 4 were found by an ILRT. The others were found by Type B and C testing or errors in test alignments. Therefore, the number of failures considered for 'small

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[9] All core damage (CD) scenario will not be impacted by the existing leakage (type A test). Examples include: the CD sequences lead to LERF due to bypass, very late vessel breach, or core rest in core, etc. Thus, the class 3 values computed with the formula are very conservative.

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releases' are 4 of 144. The EPRI Interim Guidance stated that one failure found by an ILRT was found in 38 ILRTs. Thus, the best estimate of the probability of a small leak is calculated as 5/182 = 0.027 [reference 11].

Therefore the frequency of release due to Class 3a failures is calculated as:

$$FREQ_{class3a} = PROB_{class3a} \times CDF = 0.027 \times 6.27E\text{-}05/yr = 1.69E\text{-}06/yr$$

Class 4:

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

Class 5:

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

Class 6:

This group is similar to Class 2, and addresses additional failure modes not typically modeled in PSAs due to the low probability of occurrence. The low failure probabilities are based on the need for multiple failures, the presence of automatic closure signals, and control room indication. Based on the purpose of this calculation, and the fact that this failure class is not impacted by Type A testing, no further evaluation is needed. This is consistent with the EPRI guidance. However, in order to maintain consistency with the previously approved methodology (i.e.– $PROB_{class6} > 0$), a conservative screening value of 1.0E-03 will be used to evaluate this class[10].

$$FREQ_{class6} = (\text{screening value}) \times CDF = 1.00E\text{-}03 \times 6.27E\text{-}05/yr = 6.27E\text{-}08/yr$$

Class 1:

Although the frequency of this class is not directly impacted by Type A testing, the SGS IPE did not model Class 3 or Class 6 type failures, and the frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3a, Class 3b, and Class 6 in order to preserve the total CDF. The revised Class 1 frequency is therefore:

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[10] Reference [11] suggested not including this category. The overall conclusion is not sensitive to this class.

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$FREQ_{class1} = FREQ_{PSAclass1} - (FREQ_{class3a} + FREQ_{class3b} + FREQ_{class6})$

$FREQ_{class1} = 2.94E\text{-}05 - (1.69E\text{-}06 + 1.69E\text{-}07 + 6.27E\text{-}08) = 2.75E\text{-}05/yr$

Class 2:

The frequency of Class 2 is the sum of those release categories identified in Table 5 as Class 2.

$FREQ_{class2} = 1.88E\text{-}06/yr$

Class 7:

The frequency of Class 7 is the sum of those release categories identified in Table 5 as Class 7.

$FREQ_{class7} = 3.05E\text{-}05/yr$

Class 8:

The frequency of Class 8 is the sum of those release categories identified in Table 5 as Class 8.

$FREQ_{class8} = 8.98E\text{-}07/yr$

Table 6 summarizes the above information by the EPRI defined classes. This table also presents exposures using the results of the SGS Level 3 analysis or the IP3 assumed La multipliers. For the Level 3 exposures, the weighted average[11] was used for each EPRI classification.

Table 6 Release Data Summarized by EPRI Class

Class	Description	Frequency (per year)	Person-Rem (Level 3)	Person-Rem (La Multiplier)
1	No Containment Failure	2.75E-05	3.34E03	
2	Large Containment Isolation Failures (failure to close)	1.88E-06	3.46E06	

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[11] The weighted average is the summation of the person-rem for the class divided by the total frequency of the class. An alternative approach is to use the largest release for the class. If we use the largest release, for instance, the class 7 will be overweighted results in a big total release. The changes in Class 3a and Class 3b will be masked. Thus, the weighted average is considered a better measurement.

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3a*	Small Isolation Failures (Type A test)	1.69E-06		3.34E04
3b*	Large Isolation Failures (Type A test)	1.69E-07		1.17E05
4	Small Isolation Failures - failure-to-seal (Type B test)			
5	Small Isolation Failures - failure-to-seal (Type C test)			
6	Other Isolation Failures (dependent failures)	6.27E-08		1.17E05
7	Failure Induced by Phenomena (Early and late failures)	3.05E-05	9.62E05	
8	Containment Bypasses (SGTR)	8.98E-07	1.65E06	
CDF	All Classes	6.27E-5		

Based on the above table, it can be seen that the SGS Level 3 results do not contain specific dose results for Classes 3a, 3b, and Class 6. Therefore the dose factors for these classes from the previously approved methodology (see Table 2) will be applied for this calculation. The class 3a equals 10 times the class 1 release and classes 3b and 6 each equals 35 times class 1 release.

Table 7 presents the person-rem frequency data determined by multiplying the frequency for each failure class by the corresponding exposure.

Table 7 Baseline Mean Consequence Measures for 3-Year Test Interval

Class	Description	Frequency (per year)	Person-Rem	Person-Rem per year
1	No Containment Failure	2.75E-05	3.34E+03	9.18E-02
2	Large Containment Isolation Failures (failure to close)	1.88E-06	3.46E+06	6.51E-00
3a*	Small Isolation Failures (Type A test)	1.69E-06	3.34E+04	5.66E-02
3b*	Large Isolation Failures (Type A test)	1.69E-07	1.17E+05	1.98E-02
4	Small Isolation Failures - failure-to-seal (Type B test)			
5	Small Isolation Failures - failure-to-seal (Type C test)			
6	Other Isolation Failures (dependent failures)	6.27E-08	1.17E+05	7.33E-03
7	Failure Induced by Phenomena (Early and late failures)	3.05E-05	9.62E+05	2.94E+01
8	Containment Bypasses (SGTR)	8.98E-07	1.65E+06	1.48E+00
CDF	All Classes	**6.27E-05**		37.54

The percent Risk Contribution due to release classes affected by the Type A Test interval is as follows:

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

Where:

$Class3a_{BASE}$ = Class 3a person-rem/year = 0.0566 person-rem/year

$Class3b_{BASE}$ = Class 3b person-rem/year = 0.0198 person-rem/year

$Total_{BASE}$ = total person-rem year for baseline interval = 37.54 person-rem/year

$\%Risk_{BASE} = [(0.0566+0.0198) /37.54] \times 100 = 0.20\%$

Step 4: Calculate the Type A leakage estimate to address the current inspection interval

The current surveillance testing requirements as proposed in NEI 94-01 [5] for Type A testing and allowed by 10 CFR 50, Appendix J [1] is at least once per 10 years based on an acceptable performance history (defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than 1.0La).

According to NUREG-1493 [7], extending the Type A ILRT interval from 3 in 10 years to 1 in 10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months. The average time for undetection is calculated by multiplying the test interval by 0.5 and multiplying by 12 to convert from "years" to "months." The recent EPRI Guidance suggested use the factor of 3.33 (60/18) to estimate the increase of Class 3b population dose increase. This is very conservative and will be used here for population dose calculation. The ASME required visual inspection (IWL) on liner will likely to detect the large liner breach (3b). For small liner breaches (3a), the likelihood of detection from the visual inspection is probably low.

Risk Impact Due to 10-year Test Interval

Based on the previously approved methodology [2,3], the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences. Using the EPRI Guidance for a 10-year interval, there is a factor of 3.33 increase in the overall probability of leakage. The results of this calculation are presented in Table 8 below. As with the baseline case, the IPE frequency of Class 1 has been reduced by the frequency of Class 3a, 3b, and Class 6 in order to preserve total CDF

Table 8 Mean Consequence Measures for 10-Year Test Interval

Class	Description	Frequency (per year)	Person-Rem	Person-Rem per year
1	No Containment Failure	2.32E-05	3.34E+03	7.73E-02
2	Large Containment Isolation Failures (failure to close)	1.88E-06	3.46E+06	6.51E-00
3a*	Small Isolation Failures (Type A test)	5.64E-06	3.34E+04	1.88E-01
3b*	Large Isolation Failures (Type A test)	5.64E-07	1.17E+05	6.59E-02
4	Small Isolation Failures - failure-to-seal (Type B test)			
5	Small Isolation Failures - failure-to-seal (Type C test)			
6	Other Isolation Failures (dependent failures)	6.27E-08	1.17E+05	7.33E-03
7	Failure Induced by Phenomena (Early and late failures)	3.05E-05	9.62E+05	2.94E+01
8	Containment Bypasses (SGTR)	8.98E-07	1.65E+06	1.48E+00
CDF	All Classes	**6.27E-05**		**37.71**

Using the same methods as for the baseline, and using the data in Table 8, the percent Risk Contribution due to release classes affected by the Type A Test interval is as follows:

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

Where:

$Class3a_{10}$ = Class 3a person-rem/year = 0.188 person-rem/year

$Class3b_{10}$ = Class 3b person-rem/year = 0.0659 person-rem/year

$Total_{10}$ = total person-rem year for baseline interval = 37.71 person-rem/year

$\%Risk_{10} = [(0.188 + 0.0659) / 37.71] \times 100 = 0.68\%$

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

$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100$

Where: $Total_{BASE}$ = total person-rem/year for baseline interval = 37.54 person-rem/year

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Total_{10} = total person-rem/year for 10-year interval = 37.71 person-rem/year

$\Delta\%\text{Risk}_{10} = [(37.71 - 37.54) / 37.54] \times 100 = 0.44\%$

Step 5: Calculate the Type A leakage estimate to address extended inspection intervals

Risk Impact due to 15-year Test Interval

If the test interval is extended to 1 in 15 years, the mean time that a leak detectable only by an ILRT test goes undetected increases to 90 months (0.5 * 15 * 12). The reference 11 suggested to use a factor of 5 (90/18) to account for the increased likelihood of fail to detect, which will be implemented here. As discussed previously, the factor of 5 is very conservative. The ASME required visual inspection (IWE) on liner likely to detect the large liner breach (Class 3b). The results for this calculation are presented in Table 9.

As with the baseline case, the PSA frequency of Class 1 has been reduced by the frequency of Class 3a, 3b, and Class 6 in order to preserve total CDF.

Table 9 Mean Consequence Measures for 15-Year test Interval

Class	Description	Frequency (per year)	Person-Rem	Person-Rem per year
1	No Containment Failure	2.00E-05	3.34E+03	6.69E-02
2	Large Containment Isolation Failures (failure to close)	1.88E-06	3.46E+06	6.51E-00
3a*	Small Isolation Failures (Type A test)	8.47E-06	3.34E+04	2.83E-01
3b*	Large Isolation Failures (Type A test)	8.47E-07	1.17E+05	9.90E-02
4	Small Isolation Failures - failure-to-seal (Type B test)			
5	Small Isolation Failures - failure-to-seal (Type C test)			
6	Other Isolation Failures (dependent failures)	6.27E-08	1.17E+05	7.33E-03
7	Failure Induced by Phenomena (Early and late failures)	3.05E-05	9.62E+05	2.94E+01
8	Containment Bypasses (SGTR)	8.98E-07	1.65E+06	1.48E+00
CDF	All Classes	**6.27E-05**		**37.82**

Using the same methods as for the baseline, and the data in Table 9, the percent Risk Contribution due to release classes affected by the Type A Test interval is as follows:

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

where:

$Class3a_{15}$ = Class 3a person-rem/year = 0.283 person-rem/year

$Class3b_{15}$ = Class 3b person-rem/year = 0.0990 person-rem/year

$Total_{15}$ = total person-rem year for baseline interval = 37.82 person-rem/year

$\%Risk_{15} = [(0.283+0.0990) / 37.82] \times 100 = 1.01\%$

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

$\Delta\%Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100.0$

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 37.54 person-rem/year

$Total_{15}$ = total person-rem/year for 15-year interval = 37.82 person-rem/year

$\Delta\%Risk_{15} = [(37.82-37.54) / 37.54] \times 100.0 = 0.75\%$

Step 6: Calculate increase in risk due to extending Type A inspection intervals

Extension of interval from 10 years to 15 years

Based on the previously approved methodology [2,3], the percent increase in risk (in terms of person-rem/yr) of these associated specific classes affected by the Type A test interval is computed as follows.

$\%Risk_{10\text{-}15} = [(PER\text{-}REM_{15} - PER\text{-}REM_{10}) / PER\text{-}REM10] \times 100$

where:

$PER\text{-}REM_{10}$ = person-rem/year for ten-year interval (for classes 1, 3a, and 3b)

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= (0.0773+ 0.188 +0.0659) person-rem/yr [Table 8]
= 0.332 person-rem/yr

PER-REM$_{15}$ = person-rem/year for fifteen-year interval (for classes 1, 3a, and 3b)
= (0.0669+0.283+0.0990) person-rem/yr [Table 9]
= 0.449 person-rem/yr

%Risk$_{10\text{-}15}$ = [(0.449– 0.332) / 0.332] x 100 = 35.3%

The percent increase on the total integrated plant risk for these accident sequences is computed as follows.

%Total$_{10\text{-}15}$ = [(Total$_{15}$ - Total$_{10}$) / Total$_{10}$] x 100

where:

Total$_{10}$ = total person-rem/year for ten-year interval

= 37.71 person-rem/year [Table 8]

Total$_{15}$ = total person-rem/year for fifteen-year interval

= 37.82 person-rem/year [Table 9]

% Total$_{10\text{-}15}$ = [(37.82–37.71) /37.71] x 100 = 0.31%

The percent increase on the total integrated plant risk from the baseline of three years for these accident sequences is computed as follows.

%Total$_{3\text{-}15}$ = [(Total$_{15}$ – Total$_{3}$) / Total$_{3}$] x 100

where:

Total$_{3}$ = total person-rem/year for ten-year interval

= 37.54 person-rem/year [Table 7]

Total$_{15}$ = total person-rem/year for fifteen-year interval

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= 37.82 person-rem/year [Table 9]

% $\text{Total}_{3\text{-}15}$ = [(37.82–37.54) /37.54] x 100 = 0.75%

Step 7: Calculate the change in risk in terms of Large Early Release Frequency (LERF)

The risk impact associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from containment could in fact result in a larger release, due to failure to detect a pre-existing leak during the relaxation period. Based on the previously approved methodology [2,3], only Class 3 sequences have the potential to result in large releases if a pre-existing leak (related with Type A test) is present. Based on reference [13], additional sequences that are not associated with the LERF due to a Type A containment leakage path include

1. Predominant release path does not go through the containment, e.g., steam generator tube rupture initiators and induced failures
2. Releases that would not meet the criteria for early releases
3. Release scrubbing that would prevent a large release despite the presence of a pre-existing leak. Such releases could include a successful actuation of containment sprays

Salem IPE divides release into four groups:

1. Large, early containment failures and large bypasses
2. Small, early containment failures and bypasses
3. Late containment failure
4. Containment intact

The first two groups are LERF contributors that will not be impacted by Type A test. The 4th group contains two categories. The first one is no core melt or no release. The second one is vessel failures at a very late stage (>6 hours). Therefore, this group is not impacted by the Type A test.

There are four releases in the 3rd group. The first one is "large, late containment failure with direct release." With no additional information, this one will be impacted by the Type A test. The second one is "large, late containment failure with subsoil release." Based on the description provided in Salem IPE, this release is not likely to be impacted by the Type A test. As stated in section 4.9.3.2 of Salem IPE, to become airborne, radionuclide must traverse about 50 feet radially and about 40 feet upward to reach plant grade. To be on the conservative side, 50% of the frequency of this release is impacted by the Type A test. The third release is "small, controlled late

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containment failure." This one is judged to be impacted by the Type A test. The fourth release is "very late containment basement melt through." Given the insignificant frequency, this release is included as to be impacted by the Type A test.

For SGS, the core damage sequences that will be impacted by the Type A test are about 2.35E-5/year. [Section 4.10 of SGS IPE contains more detailed information]. Using the LERF equation suggested in Ref [13]

> Frequency 3b=(3b Failure probability)*(CDF minus CDF with independent LERF minus CDF that cannot cause LERF)

Thus, the base LERF value (related with changes of ILRT interval] is equal to:

$$LERF_{baseline} = 2.35E\text{-}5 \times 0.0027 = 6.33E\text{-}8/year$$

The visual inspection (IWE) will very much likely to detect large liner failures. The IWE is scheduled based on the ASME table [12]. The percentage of containment inspected each time and the frequency depends on containment's life. For the cycle and the extension period we are in, these IWEs will cover 100% of the containment. Thus, the timed average coverage is 50% and will be credited in this analysis. Thus, the likelihood of detect a large liner failure is the same as failure to detect it (50-50 chance), the LERFs for 10 years[12] and 15 years test intervals are:

$$LERF_{10year} = LERF_{baseline} \times 3.333 \times 0.5 = 1.06E\text{-}7$$

$$LERF15year = LERF_{baseline} \times 5.00 \times 0.5 = 1.5\ 8E\text{-}7$$

Thus, the estimation for LERF changes from the 10 year interval to the 15 year test interval is 5.28E-08/year. Similarly, the LERF changes from the 3 year interval to the 15 year test interval is 9.50E-08/year. The following table summarizes the results:

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[12] The LERF for 10 year interval is under estimated with this method since it can credit a value less than 0.5. Thus, the delta LERF from 10 to 15 years is conservative.

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Table 10 Change in LERF Due to Extending Type A testing Intervals

	3-Year Interval (baseline)	**10-Year Interval**	**15-Year Interval**
Type A LERF (Class 3b)	6.33E-08/yr	1.06E-07/yr	1.58E-07/yr
ΔLERF (10- to 15-year interval)			5.28E-08/yr
ΔLERF (3-- to 15-year interval)			9.50E-08/yr

Reg. Guide 1.174 [8] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr. Since the ILRT does not impact CDF, the relevant metric is LERF. As indicated by the above table, increasing the ILRT interval from 10 to 15 years (5.28E-08/yr) is non-risk-significant. In addition, increasing the ILRT interval from 3 to 15 years (9.50E-08/yr) is non-risk-significant.

Step 8: Calculate the change in Conditional Containment Failure Probability (CCFP)

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

$$CCFP = 1 - \left[\frac{f(ncf)}{CDF} \right]$$

Where f(ncf) is the frequency of those sequences which result in no containment failure (ncf). This frequency is determined by summing the Class 1 and Class 3a results, and CDF is the total frequency of all core damage sequences.

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

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$$\Delta CCFP_{10-15} = \left[\frac{f_{Class1} + f_{Class3a}}{CDF}\right]_{10} - \left[\frac{f_{Class1} + f_{Class3a}}{CDF}\right]_{15}$$

Using the data from Table 8 and Table 9:

$$\Delta CCFP_{10-15} = \left[\frac{(2.32E-05)+(5.64E-06)}{6.27E-05}\right]_{10} - \left[\frac{(2.00E-05)+(8.47E-06)}{6.27E-05}\right]_{15}$$

$$\Delta CCFP_{10-15} = .0045 = 0.45\%$$

Using the data from Table 7 and Table 9 provide the change in CCFP from the baseline case:

$$\Delta CCFP_{3-15} = \left[\frac{(2.75E-05)+(1.69E-06)}{6.27E-05}\right]_{3} - \left[\frac{(2.00E-05)+(8.47E-06)}{6.27E-05}\right]_{15}$$

$$\Delta CCFP_{3-15} = .0108 = 1.08\%$$

6.0 RESULTS

The specific results are summarized in Table 11 below. The Type A contribution to LERF is calculated in Step 7. Based on the data:

1. The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current once every 10 years to once every 15 years is 0.11 person-rem/yr.
2. The total integrated increase in risk contribution from extending the ILRT test frequency from the current once every 10 years to once every 15 years is 0.31%.
3. The risk increase in LERF from extending the ILRT test frequency from the current once every 10 years to once every 15 years is 5.28 x 10^{-8}/yr.

The change in CCFP from the current 10-year interval to a 15-year interval is 0.45%.

Based on the above results, the following are conclusions regarding the assessment of the plant risk associated with extending the Type A ILRT test interval from ten years to fifteen years.

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The change in Type A test frequency from once every ten years to once every fifteen years increases the risk impact on the total integrated plant risk by only 0.31%. Also, the change in Type A test frequency from the original every three years to once every fifteen years increases the risk only 0.75%. Therefore, the risk impact when compared to other severe accident risks is negligible.

Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from the current once every 10 years to once every 15 years is 5.28E-8/yr. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from 10 to 15 years is therefore considered non-risk significant. The LERF increase is 9.50E-8 for the cumulative change of going from a test frequency of three every ten years to once every fifteen years, which is still non-risk significant.

R.G. 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 0.45% for the proposed change and 1.08% for the cumulative change of going from a test frequency of three every ten years to once every fifteen years. These changes are small and that the defense-in-depth philosophy is maintained.

Table 11 Summary of Risk Impact on Extending Type A ILRT Test Frequency

	Risk Impact for 3-year interval (baseline)	**Risk Impact for 10-year interval (current requirement)**	**Risk Impact for 15-year interval (proposed)**
Total Integrated Risk (Person-Rem/yr)	37.54	37.71	37.82
Type A Testing Risk (Person-Rem/yr)	0.0764	0.254	0.382
% Total Risk (Type A / Total)	0.20%	0.68%	1.01%
Type A LERF (Class 3b) (per year)	6.33E-08	1.06E-07	1.58E-07

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Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			0.11
% Increase from current (Δ Risk / Total Risk)			0.31%
Δ LERF from current (per year)			5.28E-08
Δ CCFP from current			0.45%
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			0.28
% Increase from baseline (Δ Risk / Total Risk)			0.75%
Δ LERF from baseline (per year)			9.50E-08
Δ CCFP from baseline			1.08%