VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 17, 2007

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

Serial No.	07-0802
NL&OS/ETS	R0
Docket No.	50-281
License No.	DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
SURRY POWER STATION UNIT 2
PROPOSED LICENSE AMENDMENT REQUEST
ONE-TIME FIVE-YEAR EXTENSION TO TYPE A TEST INTERVAL

Pursuant to 10 CFR 50.90, Dominion requests an amendment, in the form of a change to the Technical Specifications to Facility Operating License Number DPR-37 for Surry Power Station Unit 2. The proposed change will permit a one-time five-year exception to the ten-year frequency of the performance-based leakage rate testing program for Type A tests as required by Regulatory Guide (RG) 1.163. This one-time exception to the requirement of RG 1.163 will allow the next Type A test to be performed no later than October 26, 2015.

Attachments 1 and 2 provide an evaluation of and the risk assessment for the proposed change, respectively. The marked-up and proposed Technical Specifications pages are provided in Attachments 3 and 4, respectively.

The proposed change has been reviewed and approved by the Facility Safety Review Committee.

To permit effective outage planning, Dominion requests approval of the proposed Technical Specification change by March 31, 2009. Upon issuance, the amendment will be implemented within 60 days.

If you have any questions or require additional information, please contact Mr. Gary Miller at (804) 273-2771.

Very truly yours,

William R. Matthews
Senior Vice President – Nuclear Operations

Attachments

1. Evaluation of Proposed License Amendment
2. Probabilistic Risk Assessment
3. Marked-up Technical Specifications Page
4. Proposed Technical Specifications Page

Commitments made in this letter: None.

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by William R. Matthews, who is Senior Vice President – Nuclear Operations, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 17th day of December, 2007.

My Commission Expires: August 31, 2008.

(SEAL)

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My Commission Expires Aug 31, 2008

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Attachment 1

Evaluation of Proposed License Amendment

Surry Power Station
Unit 2
Virginia Electric and Power Company
(Dominion)

EVALUATION OF PROPOSED LICENSE AMENDMENT

1.0 INTRODUCTION

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests a change to the Surveillance Requirements in Section 4.4 of the Technical Specifications for the containment structure. The proposed change will permit a one-time five-year exception for Surry Unit 2 from the requirement of Regulatory Guide (RG) 1.163 (Reference 3) and NEI 94-01 (Reference 1), which specify performance of a Type A test at a frequency of up to ten years with allowance for a fifteen-month extension.

2.0 PROPOSED CHANGE

This application for amendment to the Surry Technical Specifications proposes to revise the Technical Specification Surveillance Requirement in Section 4.4.B.1, Containment Leakage Rate Requirements. The exception is to allow Type A testing within fifteen years from the last Type A, performed on October 26, 2000. This application represents a cost beneficial licensing change. From a differential safety benefit perspective, the improvement by performing the integrated leak rate test within ten years rather than fifteen years is not commensurate with the significant additional cost associated with the test frequency. The specific change to TS Section 4.4.B.1 (page 4.4.1) is as follows:

- Revise the current Unit 1 exception from NEI 94-01 from Unit 1 to Unit 2:

  "NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after the October 26, 2000 Type A test shall be performed no later than October 26, 2015."

3.0 BACKGROUND

3.1 10 CFR 50, Appendix J, Option B Requirements

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the technical specifications. The limitation of containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident.

The adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but it did alter the frequency of measuring primary containment leakage in Type A, B and C tests. Frequency is based upon an evaluation which looks at the "as found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The changes to the Type A test frequency did not result in an increase in containment leakage. Similarly, this proposed change to the Type A test frequency will not result in an increase in containment leakage.

3.2 Reason for Proposed Amendment

The frequency interval for testing allowed by RG 1.163 and NEI 94-01 is based upon a generic evaluation documented in NUREG-1493 (Reference 5). NUREG-1493 made the following observations with regard to extending the test frequency:

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

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

Exceptions to the requirements of RG 1.163 and NEI 94-01, are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states, "The Regulatory Guide or other implementing document used by a licensee or applicant for an operating license, 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." Since exceptions meeting the stated requirements are permitted, license amendment requests satisfying these requirements do not require an exemption to Option B.

The Surry Power Station Unit 2 current ten-year Type A test interval ends on October 26, 2010. In order to meet the interval requirements of NEI 94-01, this test must be performed during refueling outage (S2-R-22), scheduled to commence in October of 2009. By granting the proposed one-time exception, Surry would benefit by not having to perform the Type A test for an additional five years. Direct cost savings are estimated at $200,000 for equipment and materials and $122,500 in expended man hours for containment setup, valve lineups, etc. as a result of elimination of the actual performance of the test. In addition, approximately 35 hours of critical path outage time can be eliminated by not performing the Type A test. The critical path time is estimated at a savings of $1.3 million.

4.0 TECHNICAL ANALYSIS

4.1 Implementing 10 CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the Technical Specifications. The limitation of containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident.

10 CFR 50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under Option A “Prescriptive Requirements” or Option B “Performance-Based Requirements.” Amendments 208/208 (Reference 2) were issued to Surry Power Station Units 1 and 2 to permit implementation of 10 CFR 50, Appendix J, Option B. Amendments 208/208 modified Technical Specification Section 4.4 to require testing in accordance with the Containment Leakage Rate Testing Program and RG 1.163 (Reference 3), respectively. RG 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8–1994 (Reference 4), subject to several regulatory positions in the guide.

Exceptions to the requirements of RG 1.163, are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states "The Regulatory Guide or other implementing document used by a licensee, or applicant for an operating license, 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." Therefore, this application does not require an exemption to Option B.

The surveillance frequency for Type A testing in NEI 94-01 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 $1.0L_a$ and performance factors were considered consistent with NEI 94-01). Based on the May 1991 and October 2000 Type A tests, the current interval for Surry Unit 2 is once every ten years.

The extension requested for Surry Unit 2 only applies to the 10 CFR 50, Appendix J, Type A test. Appendix J, Type B and Type C tests are performed at the intervals required by Appendix J, Option B. The current rule for Type B requires completion of electrical penetrations within 120 months. Some portion of other required Type B tests are conducted each refueling, and are completed in approximately 60-month intervals consistent with the Type C testing requirements.

4.2 Surry Type A Test History

A Type A test can detect containment leakage due to a loss of structural capability. All other sources of containment leakage detected in Type A test analyses can be detected by the Type B and C tests.

Previous Type A tests confirmed that the Surry Unit 2 reactor containment structure has low leakage and represents insignificant potential risk contributor to increased containment leakage. The increased leakage is minimized by continued Type B and Type C testing for penetrations with direct communication with containment atmosphere. Also, the In-Service Inspection (ISI) program and Maintenance Rule program require periodic inspection of the interior and exterior of the containment structure to identify degradation.

The results for the last three Type A tests are reported in the following table for Surry Unit 2.

Test Date	As-Found Leakage		Acceptance Limit (*)
November 1986	Measured Leakage with upper confidence Limit (UCL) Margin	0.635 of L_a	
	Type C Penalty	0.003 of L_a	
	TOTAL	0.638 of L_a	$1.0L_a$
May 1991	Measured Leakage with UCL Margin	0.414 of L_a	
	Type C Penalty	0.004 of L_a	
	TOTAL	0.418 of L_a	$1.0L_a$
October 2000	Measured Leakage with UCL Margin	0.050 of L_a	
	Type C Penalty	0.010 of L_a	
	TOTAL	0.060 of L_a	$1.0L_a$

* The total allowable "as-left" leakage is 0.75 L_a, (L_a, 0.1% of primary containment air by weight per day, is the leakage assumed in dose consequences) with 0.6 L_a, the maximum leakage from Type B and C components.

4.3 Description of Containment

The containment is a steel-lined, heavily reinforced concrete structure with vertical cylindrical wall and hemispherical dome supported on a flat base mat. Below grade, the containment structures are constructed inside a cofferdam of steel sheet piling. The structures are soil-supported. The base of the foundation mat is located approximately 66 feet below finished ground grade.

Each containment structure has an inside diameter of 126 ft. 0 in. The spring line of the dome is 122 ft. 1 in. above the top of the foundation mat. The inside radius of the dome is 63 ft. 0 in. The interior vertical height is 185 ft. 1 in., and the base mat is 10 ft. 0 in. thick. The steel liner for the wall is 3/8-inch thick, except over the base mat, where 0.25-inch and 0.75-inch plate is used. The steel liner for the dome is 0.50-inch thick. A waterproof membrane is placed below the containment structural mat and carried up the containment wall to ground level. Attached to and entirely enveloping the part of the structure below grade, the membrane protects the structure from the effects of ground water and the steel liner from external hydrostatic pressure. Ground water immediately adjacent to the containment structure is kept below the top surface of the foundation mat by pumping as required.

Access to the containment structure is provided by a 7 ft. inside diameter personnel hatch penetration, and a 14 ft. 6 in. inside diameter equipment hatch penetration. Other smaller containment structure penetrations include hot and cold pipes, main steam and feedwater pipes, fuel transfer tube, and electrical conductors.

The reinforced concrete structure has been designed to withstand all loadings and stresses anticipated during the operation and life of the unit. The steel lining is attached to and supported by the concrete. The liner functions primarily as a gastight membrane and transmits incident loads to the concrete. The containment structure does not require the participation of the liner as a structural component. No credit has been taken for the presence of the steel liner in designing the containment structure to resist seismic force or other design loads.

The steel wall and dome liners are protected from potential interior missiles by interior concrete shield walls. CRDM missile protection is provided by a concrete shield on Unit 1 and a steel shield on Unit 2. The base mat liner is protected by a 1.5 to 2-foot thick concrete cover, except where a 0.75-inch-thick liner plate was used beneath the reactor vessel incore instrumentation, and at a drainage trench where floor grating provides additional protection.

The design basis accident was selected as the design basis for the containment structure because all other bases would result in lower temperatures and pressures. The containment structure is also designed for the normal subatmospheric operating conditions. Further, the containment structure is designated for a leakage rate not to exceed 0.1% of the contained volume per day at 45 psig.

The operating pressure for the containment is greater than 10.1 psia and less than 11.3 psia partial air pressure for Unit 2. The temperature of the containment air fluctuates between a maximum temperature of 125°F and a minimum of 75°F during normal operation and 60°F during shutdown, depending upon the ambient temperature of available service water. The normal operating pressure allows accessibility for inspection and minor maintenance during operation without requiring containment pressurization. The containment structure is designed by ultimate strength methods conforming to ACI 318-63, Part IV-B.

4.4 Containment Leakage Consideration for Operability

Containment operability is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to entering a Mode where operability is required for the first time following the performance of a periodic test performed in accordance with 10 CFR 50, Appendix J, Option B (see Surry TS 4.4.B.2). At that time the combined Type B and C leakage must be < 0.6 L_a on a maximum pathway leakage rate basis and the overall Type A leakage must be < 0.75 L_a. At all other times prior to performing as found testing, the acceptance criteria for Type B and C leakage testing is < 0.60 L_a on a minimum pathway leakage rate basis. In addition, to leakage considerations following a design basis LOCA, containment operability also requires structural integrity following a design basis accident.

Compliance with the Technical Specification discussed above will ensure a containment configuration, including personnel and equipment hatches, that is structurally sound and that will limit leakage to the rates assumed in the safety analysis.

4.5 Containment Operational Performance

During power operation, Surry Unit 2 is maintained at a subatmospheric condition. Instrumentation constantly monitors containment pressure. If pressure rises, an alarm annunciates conditions approaching the limits specified in the Technical Specifications. Although not as significant as the differential pressure resulting from a design basis accident, the fact that the containment can be maintained subatmospheric provides a degree of assurance of containment structural integrity (i.e., no large leak paths in the containment structure). This feature is a complement to visual inspection of the interior and exterior of the containment structure for those areas that may be inaccessible for visual examination.

4.6 IWE/IWL Inservice Inspection (ISI) Activities to Support Type A Test

Surry Unit 2 has completed the requirements of their first ten-year concrete containment inservice inspection program (IWL). Concrete containment examinations were completed by August 31, 2007 in accordance with the requirements of the 1992 Edition with the 1992 Addenda of ASME Section XI completing the first ten-year interval. These examinations on the concrete exterior were conducted by the Responsible Engineer using the visual method (VT-3C and VT-1C). The second ten-year concrete containment examinations (IWL) will be completed in accordance with the requirements of the 2001 Edition through the 2003 Addenda of ASME Section XI and have specified dates of August 31, 2011 and August 31, 2016. General and detailed visual examinations shall be completed in accordance with Category L-A of the Code no earlier than or later than one year of the specified date. The Surry Unit 2 containment does not have an unbonded post-tensioning system. As such, examinations required by Category L-B do not apply.

The first ten-year interval metallic shell and penetration liner examination (IWE) requirements are still being completed. These examinations are being in accordance with the 1992 Edition with the 1992 Addenda of ASME Section XI with the one exception for Item E1.12 (wetted surfaces of submerged areas). For this item, the NRC recently approved the request to use the 2001 Edition through the 2003 Addenda of ASME Section XI as modified by 10 CFR 50.55a(b) limitations. The interval date for Surry Unit 2 is October 20, 1997 to May 21, 2008.

Examinations performed for Surry Unit 2 include the general visual of all accessible areas, and a visual (VT-1) examination of pressure retaining bolting. Additionally, visual (VT-3) examinations are planned for item E1.12 (wetted surfaces of submerged areas). The examination requirements are detailed in Categories E-A and E-G of the applicable Code. No other examination categories apply.

The second ten-year interval IWE examination requirements will be performed in accordance with the 2001 Edition through the 2003 Addenda of ASME Section XI as modified by the 10 CFR 50.55a (b) limitations. At this time no augmented Category E-C examinations are planned. The remaining examinations are based on Category E-A, and are visual (General, VT-3, and VT-1) examinations based on Code or 10 CFR requirements.

The following relief requests were reviewed to assess the effect, if any, resulting from the proposed Type A test frequency extension:

- Relief Request RR-IWE2 obtained relief from Section XI of the ASME Code, 1992 Edition, 1992 Addenda, Code Items E5.10 and E5.20 which require a visual examination of metal containment seals and gaskets. The relief permits continued acceptance of containment seals and gaskets through the performance of 10 CFR Appendix J testing rather than by individual visual inspection. NRC letter dated April 14, 1999 granted this relief to Surry Units 1 and 2. The proposed Type A test frequency extension only affects Type A testing. The Type B testing program remains unaffected and, therefore, the relief request remains valid and unaffected by the proposed change.

- Relief Request RR-IWE5 obtained relief from Section XI of the ASME Code, 1992 Edition, 1992 Addenda, Code Item E8.20 which requires a bolt torque or tension test for bolted connections that have not been disassembled and reassembled during the inspection interval. The relief request permits the leak tightness of bolted connections to be verified through the performance of 10 CFR 50 Appendix J testing. NRC letter dated April 21, 1999 granted the relief request. The proposed frequency extension affects Type A testing only. The Type B testing program is not affected. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request RR-IWE8 obtained relief from Section XI of the ASME Code 1992 Edition, 1992 Addenda, Table IWE-2500-1, Category E-P, which contains examination requirements in conjunction with post repair, replacement, and 10 CFR 50 Appendix J requirements. NRC letter dated March 8, 2000 granted the relief request for Surry Units 1 and 2. The relief request is administrative in nature, removing redundant Code requirements addressed by Appendix J and eliminating unnecessary Authorized Nuclear Inservice Inspector (ANII) involvement. As a result, the relief request remains valid and unaffected by the proposed change.

The second ten-year interval IWE program for Surry meets the requirements of the 2001 Edition through the 2003 Addenda of ASME Section XI. Categories E-D and E-G are no longer part of the Code. The relief requests above are not needed for the second ten-year interval since examination of seals and gaskets and bolt torque or tension tests are no longer addressed by ASME Section XI. As such, the extension request will no longer impact the ASME Section XI program upon second interval start for each unit. Given the short time period remaining in the first ten-year IWE ISI interval for Surry Unit 2, and the Type B and C tests performed during the first ten-year IWE ISI interval, the Appendix J, Type A extension is seen as having a negligible impact.

Surry Engineering performs IWE/IWL ISI inspection activities in support of the required Type A test. There will be no change to the schedule for these inspections due to the extension of the Type A test interval. The activities performed that assure continued containment integrity include:

- During refueling outages, IWE general visual examinations are performed on the Containment Metal Liner (IWE - MC component). All or parts of accessible areas are examined. Although localized rust and surface anomalies were detected, no repairs have been required to meet Code requirements.

- The 2002 IWL containment ISI Program inspections of the Surry Unit 2 containment structure identified embedded material in the containment dome area. In the Unit 2 containment dome the material appears to be wood, approximately 2 in. x 12 in. with the side grain exposed. The findings represent a direct inspection of approximately one-third of the containment dome and a remote inspection of the remainder of the containment dome.

  The embedded material, as described above, was inadvertently left in the containment structure during original plant construction. The slight depression in the location of the wood and below the adjacent concrete indicates that the wood was likely concealed below a thin layer of cement paste immediately following removal of the concrete form-work. Over time this thin layer of concrete has spalled off, leaving the wood exposed.

Engineering performed an assessment of the significance of the embedded material. The assessment concluded that the containment structures remain fully capable of meeting the functional design requirements as described in Technical Specification 5.2 and UFSAR Section 15.5. This assessment assumed that the piece of wood extends from the concrete surface through the concrete placement and also assumed that similar embedded pieces of wood could exist in the two-thirds of the containment dome that was not directly inspected at that time. An Engineering evaluation of the inspection findings was performed and concluded that:

- the leak-tight integrity of the liner has not been jeopardized,
- any degradation of the underlying reinforcing steel as a result of the embedded wood is insignificant to the structure,
- the loss of concrete displaced by the wood will have an insignificant effect upon the structure, and
- no significant loss of radiological shielding or missile protection has occurred.

- Direct inspection of the remainder of the containment dome was completed in April 2002. Any identified embedded debris was removed and the containment repaired, as necessary.

The ASME Section XI, Article IWL interval 2 (2nd 5-year) inspection of the Surry Unit 2 containment structure was conducted between May 2006 and December 2006. The following was observed during the inspection:

- Embedded materials were identified in the containment dome area. The materials included small scraps of dimensional lumber, a rubber gasket, metal rod, and wire. This debris was all located within the concrete cover above the first row of steel reinforcing bar. No reinforcing bar was exposed as a result of debris removal and no concrete deterioration was identified. All identified debris was extracted and all void areas deeper than one half inch were repaired using cement based mortar.

- Numerous pop-outs and abandoned anchors were identified on the containment exterior concrete and uncoated areas of the containment structure adjacent to other building interiors. The pop-outs and abandoned anchors identified on the exterior concrete were repaired using cement based mortar.

- Rock pockets, sand pockets, a crack, a spall, two small holes, and a number of loose patches were identified. All identified defects were excavated to sound material and repaired using cement based mortar or concrete.

Based upon these inspection findings, the Unit 2 containment structure is generally found to be in good material condition. No significant defects or concerns were observed on the exterior concrete and all observed defects were due to original construction flaws. Taken together or individually, the defects identified did not represent a significant structural concern. The containment structure continues to retain its ability to perform as designed under all load cases including the design basis earthquake and a postulated strike from a tornado generated missile.

During the fall 2003 (S2-R-18) refueling outage Surry Unit 2 replaced the reactor pressure vessel head. Completion of the replacement head required an opening in the containment larger than the equipment hatch. Therefore, an opening ranging from a size of approximately 18 ft. 6 in. high by 17 ft. 0 in. wide in the steel liner plate and 26 ft. 0 in. high by 27 ft. 4 in. wide at the outside face of the concrete. The opening was repaired, examined, and tested in accordance with the appropriate ASME Code requirements for the metal liner and concrete structure.

The cut steel liner plate was welded back to its original configuration using full penetration welds. After the cut steel liner plate was welded back to its original configuration, Non-Destructive Examination (NDE) was performed on the completed full penetration weld. A surface examination was performed on the weld as well as a spot radiographic examination. In addition, although not an original design requirement, vacuum box testing was performed as required in accordance with Relief Request IWE9 (Reference 8). After vacuum box testing of the liner seam weld and installation of the channel, the channel to liner weld was tested by a static pressure test (decay test) and the weld was soap bubble tested for leakage with an acceptance criterion of zero leakage. In addition, following the containment building pressure test, the channel was pressurized and an "as-found" local leak rate test was performed at 50 psi in accordance with NRC approved relief request RR-IWE9.

In accordance with IWL Article 5000 of ASME Code, a containment structure pressure test was performed at 45 psi. The surface of the replacement concrete was examined in accordance with IWL-5250 prior to pressurization, at test pressure and following completion of the pressurization test. The extensive testing ensures that the Unit 2 containment structure has been restored to its original design condition.

5.0 PLANT SPECIFIC RISK ASSESSMENT FOR THE EXTENDED TYPE A TEST INTERVAL

5.1 Method of Analysis

A simplified bounding analysis approach was used for evaluating the change in risk associated with increasing the interval for performing the Type A test from ten years to fifteen years.

The Type A test measures the containment air mass and calculates the leakage from the change in mass over time. Likewise, this approach is used in the analyses presented in EPRI TR-104285, NUREG-1493, and the NEI Interim Guidance. The analysis performed examines plant specific accident sequences in which the containment integrity remains intact or the containment is impaired. Specifically, the following were considered:

- 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 (e.g., a liner breach or steam generator manway leakage [EPRI TR-104285 Class 3 sequences]). Type B tests measure component leakage across pressure retaining boundaries (e.g., gaskets, expansion bellows, and air locks). Type C tests measure component leakage rates across containment isolation valves.

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left opened following a plant post-maintenance test (e.g., a valve failing to close following a valve stroke test [EPRI TR-104285 Class 6 sequences]).

- Accident sequences involving containment failure induced by severe accident phenomena (EPRI TR-104285 Class 7 sequences), containment bypassed (EPRI TR-104285 Class 8 sequences), and large containment isolation failures (EPRI TR-104285 Class 2 sequences).

- Small containment isolation 'failure-to-seal' events (EPRI TR-104285 Class 4 and 5 sequences) were not accounted for in this evaluation. These sequences are impacted by changes in Type B and C test intervals, not changes in the Type A test interval.

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left 'opened' following a plant post-maintenance test (e.g., valve failing to close following a valve stroke test) where not accounted for in this evaluation (EPRI TR-104285 Class 6 sequences).

5.2 Conclusions

Based on the above sequences considered, the following conclusions are made regarding the plant risk associated with extending the Type A test frequency from ten years to fifteen years:

- RG 1.174 (Reference 8) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-5/yr and increases in large early release frequency (LERF) below 1E-6/yr. Since the Type A test does not impact CDF, the relevant criterion is LERF.

- The increase in LERF resulting from a change in the Type A test interval from once-per-ten-years to once-per-fifteen-years is 1.3E-7/yr based on internal events. RG 1.174 states that when the calculated increase in LERF is in the range of 1E-7/yr to 1E-6/yr, applications will be considered if it can be shown that the total LERF is less than 1E-5/yr. Since the total LERF for the 15-year metric is 9.8E-7/yr, then the proposed change is considered acceptable.

- The increase in the total dose rate is defined here by person-rem/year increases for those accident sequences influenced by Type A testing. The one-time change to the Type A test interval from ten years to fifteen years increases the Type A test dose rate by 0.024%. This change in dose rate is due to the conservative assumption made in the calculation of the Class 3 frequencies.

- The change in conditional containment failure probability (CCFP) is calculated to demonstrate the impact on 'defense-in-depth.' The changes in the contribution of sequences involving containment failure are negligible from the ten-year interval to the proposed fifteen-year interval. Therefore, the Surry Unit 2 containment defense-in-depth is maintained.

The risk assessment calculation performed for the 5-year Type A test extension for Unit 2 from 10 years to 15 years is included in Attachment 2.

6.0 EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

The proposed revision to Technical Specifications permits a one-time extension to the current interval for Type A testing. The current test interval of ten years, which is based on the standard of good past performance, will be extended on a one-time basis to fifteen years from the last Type A test. In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based upon the following information:

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

   The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since extension of the containment Type A testing is not a physical plant modification that could alter the probability of accident occurrence nor is an activity or modification by itself that could lead to equipment failure or accident initiation.

The proposed extension to Type A testing does not result in a significant increase in the consequences of an accident as documented in NUREG-1493. The NUREG notes that very few potential containment leakage paths are not identified by Type B and C tests. It concludes that reducing the Type A testing frequency to once per twenty years leads to an imperceptible increase in risk.

Surry provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last two Type A tests identified containment leakage within acceptance criteria, indicating a very leak-tight containment. Inspections required by the ASME Code are also performed in order to identify indications of containment degradation that could affect leak-tightness. Separately, Type B and C testing, required by Technical Specifications, identifies any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that an extension to the Surry Type A test interval will not represent a significant increase in the consequences of an accident.

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

   The proposed revision to Surry Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing does not create the possibility of a new or different type of accident since 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.

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

   The proposed revision to Surry Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, will be extended on a one-time basis to fifteen years from the last Type A test. Regulatory Guide (RG) 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines small changes in risk as resulting in increases of CDF below 1E-5/yr and increases in LERF below 1E-6/yr. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from a once-per-ten-years to a once-per-fifteen-years is 1.3E-7/yr, based on internal events. RG 1.174 states that when the calculated increase in LERF is in the range of 1E-7/yr to 1E-6/yr, applications will be considered if it can be shown that the total LERF is less than 1E-6/yr. Since the total LERF is 9.8E-7/yr, the change is considered small and not a significant reduction in margin. Increasing the Type A test interval from ten to fifteen years is, therefore, considered non-risk significant and 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 extension in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG-1493 found that, generically, the design containment leakage rate contributes about 0.1 percent of the overall risk and that decreasing the Type A testing frequency would have a minimal affect on this risk since 95% of the Type A detectable leakage paths would already be detected by Type B and C testing. Furthermore, for Surry, maintaining the containment subatmospheric during plant operations further reduces the risk of any containment leakage path going undetected.

7.0 IMPLEMENTATION OF THE PROPOSED CHANGE

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described in Section 6 of this evaluation, the proposed change involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change does not involve the installation of any new equipment, or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupation radiation exposure.

The proposed change does not involve plant physical changes, or introduce any new mode of plant operation. The extended interval will not include any activities that will increase individual or cumulative occupation radiation exposure. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above, Dominion concludes that the proposed changes meet the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.22 relative to requiring a specific environmental assessment by the Commission.

8.0 CONCLUSION

The proposed one-time change will not alter assumptions relative to the mitigation of an accident or transient event and will not adversely affect normal plant operation and testing. The proposed change is consistent with the current safety analysis assumptions and with the Technical Specifications. As such, no question of safety exists.

RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines small changes in risk as resulting in increases of CDF below 1E-5/yr and increases in LERF below 1E-6/yr. Since the Type A does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from once-per-ten-years to once-per-fifteen-years is 1.3E-7/yr, based on internal events. RG 1.174 states that when the calculated increase in LERF is in the range of 1E-7/yr to 1E-6/yr, applications will be considered if it can be shown that the total LERF is less than 1E-5/yr. Since the total LERF is 9.8E-7/yr then the change is considered acceptable.

The Facility Safety Review Committee has reviewed this proposed change to the Technical Specifications and have concluded that it does not involve a significant hazards consideration and will not endanger the health and safety of the public.

9.0 REFERENCES

1. 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.

2. NRC letter to Surry Units 1 and 2, Technical Specification Amendments 208/208, dated, April 18, 1996 that permitted implementation of the requirements of 10 CFR 50, Appendix J, Option B for performance-based primary reactor containment leakage testing.

3. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

4. American National Standard ANSVANS - 56.8 - 1994, "Containment System Leakage Testing Requirements."

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

6. EPRI TR-104285, "Risk Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994.

7. Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leak Rate Tests for Surveillance Intervals, Dated November 2001.

8. NRC letter to Surry Power Station approving RR-IWE9, "Surry Power Station, Unit 2 - ASME Section XI, Inservice Inspection Program, Relief Request (RR) RR IWE9 for Containment Testing," dated May1, 2003.

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

Attachment 2

Probabilistic Risk Assessment
Five Year Type Extension for Surry Unit 2

Surry Power Station
Unit 2
Virginia Electric and Power Company
(Dominion)

Surry Power Station
Probabilistic Risk Assessment Notebook

Part V PRA Risk Analysis

Volume RA.LI.4

Calculation of 5 year ILRT Extension for Unit 2 from 10 years to 15 years

Revision No. 2
Effective Date: December 2007

Purpose: To provide a risk impact assessment on extending the Integrated Leak Rate Test (ILRT) interval for Surry Unit 2 from 10 years to 15 years.		
Conclusion: The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once-per-fifteen-years is 1.33E-07 / yr. Therefore, the risk impact when compared to other severe accident risks is small.		
Prepared By:	**Signature**	**Date**
Reviewed By:	**Signature**	**Date**
Approved By:	**Signature**	**Date**

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SUMMARY OF CHANGES

Revision	**Author**	**Summary**
0		Initial issuance.
1		Editorial clarification of comments
2		Clarification of the Conditional Containment Failure Probability for the 15 year test interval

1.0 PURPOSE

The purpose of this document is to provide a risk impact assessment on extending the Integrated Leak Rate Test (ILRT) interval for Surry Unit 2 from once in 10 years to once in 15 years.

2.0 INTRODUCTION

On October 26, 1995, the NRC revised 10 CFR 50, Appendix J. The revision to Appendix J allowed individual plants to select containment leakage testing under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements". The Surry Nuclear Power Station (SPS) selected the requirements under Option B as its testing program [PROCDR01].

The surveillance testing requirements as proposed in NEI 94-01 [REPORT01] for Type A testing is at least once every 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 $1L_a$).

The Surry Unit 2 current 10-year Type A test interval ends in October 2010. The proposed amendment to the Technical Specification (TS) is for a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A test as documented in [REPORT01]. The exception will allow ILRT testing within 15 years from the last ILRT which was performed in October 2000.

This calculation will provide a risk impact assessment on extending the plant's ILRT interval by five years. The risk assessment will be performed in accordance with the guidelines set forth by NEI [REPORT01] and [REPORT02], the methodology used by EPRI [REPORT03] and [REPORT04], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, Regulatory Guide 1.174 [RG01].

In addition, the results and findings from the Surry Individual Plant Examination (IPE) [REPORT09], the revised model [CALC02], [CALC04], [CALC11], and previous calculations [CALC06], [CALC07], and [CALC09] were used for this risk assessment calculation.

3.0 ANALYSIS

3.1 Inputs

This calculation will use Surry 50 mile population data for calculating the population dose, which was also used for license extension Severe Accident Mitigation Alternatives (SAMA) analysis as discussed in [CALC01]. The Source Term Category (STC) release fractions and corresponding frequencies were taken from the [CALC11] and revised data in [CALC02]. Source term category is defined here as a grouping of like releases from the Containment Event Tree (CET) endpoints such that the offsite consequences are expected to be similar. There are enough STCs to cover the spectrum of releases.

3.2 Assumptions

As stated in the Surry Technical Specifications, the leakage rate (L_a) acceptance criterion is defined as:

L_a = 0.1 percent by weight of containment air per 24 hours at calculated peak pressure (P_a)

The NEI interim guidance [REPORT02] was instrumental in making all of the following assumptions:

1. Containment leak rates greater than $1L_a$, but less than $35L_a$, indicate an impaired containment. Leak rates within this range are considered 'small'.

2. Containment leak rates greater than $35L_a$, indicate a containment breach. These leak rates are considered to be 'large'.

3. Containment leak rates less than $1L_a$ indicate an intact containment. These leak rates are considered to be 'negligible'.

4. The maximum containment leakage for Class 3A sequences is $10L_a$.

5. The maximum containment leakage for Class 3B sequences is $35L_a$.

6. 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.

7. Containment leakage related to Classes 4, 5, and 6 are not affected by changes in ILRT test frequency. Therefore, these classes are not considered in this assessment methodology.

8. The containment releases for Classes 2, 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 a Type A.

9. The containment releases are not impacted with time.

3.3 Methodology

A simplified bounding analysis approach for evaluating the change in risk associated with increasing the interval from 10 years to 15 years for the Type A test was used. Type A tests measure the containment air mass and calculates the leakage from the change in mass over time. This approach is similar to that presented in the EPRI [REPORT03] and NEI [REPORT02] reports, as well as NUREG-1493 [NUREG01]. Namely, the analysis performed examined the SPS IPE [REPORT09] plant specific accident sequences in which the containment integrity remains intact or the containment is impaired. Specifically, the following were considered:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR104285 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 steam generator man way leakage. (EPRI TR-104285 Class 3 sequences). Type B test measures component leakage across pressure retaining boundaries (e.g. gaskets, expansion bellows and air locks). Type C test measures component leakage rates across containment isolation valves.

- Accident sequences involving containment failure induced by severe accident phenomena (EPRI TR-104285 Class 7 sequences), containment bypassed (EPRI TR-104285 Class 8 sequences) and large containment isolation failures (EPRI TR-104285 Class 2 sequences).

- Small containment isolation 'failure-to-seal' events (EPRI TR-104285 Class 4 and 5 sequences) were not accounted for in this evaluation. These sequences are impacted by changes in Type B and C test intervals, not changes in the Type A test interval.

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left 'opened' following a plant post-maintenance test (e.g. a valve failing to close following a valve stroke test) where not accounted for in this evaluation (EPRI TR-104285 Class 6 sequences).

<u>The steps taken to perform this risk assessment evaluation are as follows:</u>

Step 1 - Quantify the baseline risk in terms of core damage frequency per reactor year for each of the eight accident classes presented in Table 1. Map the Level 3 release categories into 8 release classes defined by EPRI TR-104285 [REPORT03]. See Table A-1 of Attachment A.

Step 2 - Develop baseline plant specific population dose in person-rem per reactor year for each of the eight accident classes evaluated in the EPRI Report [REPORT03].

Step 3 - Evaluate risk impact of extending the Type A test interval from 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 [RG01]

Step 5 – Evaluate the Risk Impact in Terms of ΔLERF

Step 6 – Determine Impact on Conditional Containment Failure Probability

4.0 BODY OF CALCULATION

Step 1 - Quantify the baseline risk in terms of core damage frequency per reactor year.

This step involves the review of the SPS IPE [REPORT09] containment event tree (CET). The CET characterizes the response of the containment to important severe accident sequences. The CET used in this evaluation is based on important phenomena and systems-related events identified in NUREG-1335 [NUREG02] and NSAC-159, Volume 2 [REPORT05] and on plant features that influence the phenomena.

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, containment failure induced by severe accident phenomena, or accidents in which containment is bypassed. As a result, the CET containment isolation model was reviewed for applicable isolation failures and their impact on the overall plant risk.

A review of the containment isolation model was performed [CALC03] and [CALC04]. The five issues associated with containment isolation in NUREG-1335 [NUREG02] were examined, and these issues are:

(1) The identity of pathways that could significantly contribute to containment isolation failure.
(2) The signals required to automatically isolate the containment penetration.
(3) The potential generating signals for all initiating events.
(4) The examination of testing and maintenance procedures.
(5) The quantification of each containment isolation mode.

The containment isolation model in [CALC03] and [CALC04] screened out lines less than 5.5 inches in diameter which was the minimum cutoff for the LERF definition. This evaluation considers lines that are sized between 0.1 inches and 5.5 inches as potential candidates for significant containment leakage.

The Level 3 release categories were mapped into 8 release classes (See Table A-1 in Attachment A) as defined in the EPRI Report [REPORT03]. These EPRI containment failure classifications are listed below.

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 L_a, under Appendix J for that plant. The allowable leakage rates (L_a), are typically 0.1 weight percent of containment volume per day for PWRs (e.g. SPS measured at Pa, 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 requirement or verified by in service inspection and testing (ISVIST) 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.

The frequencies for the above eight classes are calculated below. The Class 3 frequencies are needed to determine the Class 1 frequencies and will be calculated first.

Class 3 Sequences: This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (i.e. containment liner) exists. The containment leakage for these sequences can be either small ($1L_a$ to $35L_a$) or large ($>35L_a$).

To calculate the probability that a liner leak will be large (Event CLASS-3B), the data presented in NUREG-1493 [NUREG01] was used. The data found in NUREG-1493 states that 144 ILRTs were conducted. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Since 21L_a does not constitute a large release (please refer to the write-up in Step 4 and Table 6 for Large Release information), no large releases have occurred based on the 144 ILRTs reported in NUREG-1493 [NUREG01].

An improvement in the methodology used to determine the frequencies of leakages detectable only by ILRTs, classes 3A and 3B was made using the methods documented in [REPORT02]. The method utilized in the aforementioned utility submittals (discussed in [REPORT02]) involved using a 95% confidence of a $\chi 2$ distribution of the noted ILRT failures (4 of 144 reported in [NUREG01]). Data collected recently by NEI from 91 nuclear power plants indicates that 38 plants have conducted ILRTs since 1/1/95, with only one failure (due to construction debris from a penetration modification). This would indicate that the statistical information should be based on 5/182. Rather than using the $\chi 2$ distribution used previously, it has been considered more appropriate to utilize the mean (5/182 = 0.027) for the class 3A (small leak) distribution. From the NEI document [REPORT02], the Jeffrey's non-informative prior distribution was used to calculate the class 3B (large leak) distribution as follows:

$$Failure\ \Pr obability = \frac{(NumberofFailures) + (0.5)}{(NumberofTests) + 1}$$

$$Failure\ \Pr obability = \frac{(0) + (0.5)}{(182) + 1}$$

The number of large failures is zero, so the class 3B probability is 0.5 / 183 = 0.0027

The respective frequencies per year are determined as follows:

CLASS-3A-FREQUENCY = $PROB_{class\text{-}3A}$ * CDF
CLASS-3B-FREQUENCY = $PROB_{class\text{-}3B}$ * CDF

<u>where:</u>
$PROB_{class\text{-}3A}$ = probability of small pre-existing containment liner leakage
= 0.027

$PROB_{class\text{-}3B}$ = probability of large pre-existing containment liner leakage
= 0.0027

CDF = 2.93E-05 / year [Table A-1, Attachment A]

CLASS-3A-Base-Frequency = 0.027 * 2.93E-05 / year = 7.91E-07 / year

CLASS-3B-Base-Frequency = 0.0027 * 2.93E-05 / year = 7.91E-08 / year

For this analysis the associated maximum containment leakage for class 3A is $10L_a$ and for class 3B is $35L_a$

Class 1 Sequences: This group consists of all core damage accident progression bins for which the containment remains intact. The frequency per year for these sequences is 3.17E-06 / year (see Attachment A, Table A-1). For this analysis the associated maximum containment leakage for this group is $1L_a$. The SPS IPE did not model Class 3 type failures; therefore they need to be accounted for in the Class 1 accident class. Using NEI interim guidance methodology [REPORT02], the frequency for Class 1 should be reduced by the new estimated frequencies in Class 3A and Class 3B in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$$\text{CLASS-1-FREQ} = \text{FREQ}_{\text{Class-1}} - (\text{FREQ}_{\text{Class3A}} + \text{FREQ}_{\text{Class3B}})$$
$$\text{CLASS-1-FREQ} = 3.17\text{E-}06 - (7.91\text{E-}07 + 7.91\text{E-}08)$$

CLASS-1-Base-Frequency = 2.30E-06 / year

Class 2 Sequences: This group consists of all core damage accident progression bins for which a pre-existing leakage due to failure to isolate the containment occurs. The frequency for Class 2 is the sum of those release categories identified in Table A-1, Attachment A as Class 2.

CLASS-2-FREQUENCY = 6.52E-10 / year

Class 4 Sequences: This group consists of all core damage accident progression bins 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 probability will not be impacted by Type A testing. Because these failures are detected by Type B tests, this group is not evaluated any further, consistent with approved methodology.

Class 5 Sequences: This group consists of all core damage accident progression bins 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 probability will not be impacted by Type A testing. Because these failures are detected by Type C tests, this group is not evaluated any further, consistent with approved methodology.

Class 6 Sequences: This group is similar to Class 2 and addresses additional failure modes not typically modeled in PRAs due to the low probability of occurrence. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution.

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.

Class 7 Sequences: This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (Early and Late Failures). The frequency of Class 7 is the sum of those release categories identified in Table A-1, Attachment A as Class 7.

CLASS-7-FREQUENCY = 2.36E-05 / year

Class 8 Sequences: This group consists of all core damage accident progression bins in which containment bypass occurs. The frequency of Class 8 is the sum of those release categories identified in Table A-1 as Class 8.

CLASS-8-FREQUENCY = 2.49E-06 / year

Note: for this class the maximum release is not based on normal containment leakage, because the releases are released directly to the environment. Therefore, the containment structure will not impact the release magnitude.

Table 1: Baseline Containment Frequencies - Given Accident Class

Class	Description	Frequency (per Rx-year)
1	No Containment Failure	2.30E-06
2	Large Containment Isolation Failures (Failure-to-close)	6.52E-10
3A	Small Isolation Failures (Type A test)	7.91E-07
3B	Large Isolation Failures (Type A test)	7.91E-08
4	Small isolation failure - failure-to-seal (Type B test)	Not Analyzed
5	Small isolation failure - failure-to-seal (Type C test)	Not Analyzed
6	Containment Isolation Failures (dependent failures, personnel errors)	Not Analyzed
7	Severe Accident Phenomena Induced Failure (Early and late Failures)	2.36E-05
8	Containment Bypassed (SGTR & V-Sequence)	2.49E-06
CDF	Core Damage All CET End states	**2.93E-05**

Step 2 – Develop baseline plant specific person-rem dose (population dose) per reactor year.

Plant-specific MAAP/MACCS2 analysis was performed to evaluate the person-rem dose to the population, within a 50-mile radius from the Surry power plant. The dose for Class 1 and Class 2 accidents is the sum of the Class 1 and Class 2 dose values from Table A-1, Attachment A, respectively.

Using the total population dose for Class 1 accidents as the starting reference point, the Class 3, Class 7, and Class 8 accidents are calculated below. The population dose is converted to the corresponding Class value using the appropriate dose multiplier as was used in the NEI methodology [REPORT02] to predict the person-rem dose for Class 3 accidents. Note that the multiplier (i.e. 10) for Class 3A is the maximum containment leakage multiplier assumed for small leaks, and the Class 3B multiplier (i.e. 35) is the maximum containment leakage multiplier assumed for large leaks. The dose for the Class 7 accidents was obtained by frequency weighting all the Class 7 dose values. This was done by dividing the sum of the products by the sum of the frequencies from Table A-1. Class 8 sequences include containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are expected to be released directly to the environment. The class 8 doses are frequency weighted as were done for Class 7. The frequency weighted Class 8 dose from Table A-1 represent the sum of the dose for the Event-V and SGTR sequences. The baseline dose results are calculated below and are summarized in Table 2.

Class 1 = 5.98E+02 person-rem
Class 2 = 2.38E+04 person-rem
Class 3A = 5.98E+02 * 10 = 5.98E+03 person-rem
Class 3B = 5.98E+02 * 35 = 2.09E+04 person-rem
Class 4 = Not analyzed
Class 5 = Not analyzed
Class 6 = Not analyzed
Class 7 = $\sum^n$ (Freq x Dose) / $\sum^n$ Freq = 1.00E+05 person-rem
Class 8 = $\sum^n$ (Freq x Dose) / $\sum^n$ Freq = 3.61E+06 person-rem

Table 2: Person-Rem Measures - Given Accident Class

Class	Description	Person-Rem (50-Miles)
1	No Containment Failure	5.98E+02
2	Large Containment Isolation Failures (Failure-to-close)	2.38E+04
3A	Small Isolation Failures (Type A test)	5.98E+03
3B	Large Isolation Failures (Type A test)	2.09E+04
4	Small isolation failure - failure-to-seal (Type B test)	N/A
5	Small isolation failure - failure-to-seal (Type C test)	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	N/A
7	Failure Induced by Phenomena (Early and Late Failures)	1.00E+05
8	Containment Bypassed (SGTR & V-Sequence)	3.61E+06

The above dose results when combined with the frequency results presented in Table 1 yields the SPS baseline mean consequence measures for each accident class. These results are presented in Table 3 below.

Table 3: Baseline Mean Person-Rem Measures - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Person-Rem (50-Miles)	Person-Rem/yr (50-Miles)
1	No Containment Failure	2.30E-06	5.98E+02	1.38E-03
2	Large Isolation Failures (Failure-to-close)	6.52E-10	2.38E+04	1.55E-05
3A	Small Isolation Failures (Type A test)	7.91E-07	5.98E+03	4.73E-03
3B	Large Isolation Failures (Type A test)	7.91E-08	2.09E+04	1.65E-03
4	Small isolation Failure-to-Seal (Type B test)	N/A	N/A	N/A

5	Small isolation Failure-to-Seal (Type C test)	N/A	N/A	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	N/A	N/A	N/A
7	Failure Induced by Phenomena (Early and Late Failures)	2.36E-05	1.00E+05	2.36
8	Containment Bypassed (SGTR & V-Sequence)	2.49E-06	3.61E+06	8.99
CDF	All CET End States	**2.93E-05**	N/A	**11.36**

Based on the above values, using the same methodology as [CALC10], the baseline percent risk contribution of the Dose Rate (DR) related to Type A testing is as follows:

% of Total DR_{BASE} = [($CLASS3A_{BASE}$ + $CLASS3B_{BASE}$) / $Total_{BASE}$] X 100

<u>where</u>:
$CLASS3A_{BASE}$ = class 3A person-rem/year = 4.73E-03 person-rem/year [Table 3]

$CLASS3B_{BASE}$ = class 3B person-rem/year = 1.65E-03 person-rem/year [Table 3]

$Total_{BASE}$ = total person-rem/year for baseline interval = 11.36 person-rem/year [Table 3]

% of Total DR_{BASE} = [($CLASS3A_{BASE}$ + $CLASS3B_{BASE}$) / $Total_{BASE}$] x 100%
% of Total DR_{BASE} = [(4.73E-03 + 1.65E-03) / 11.36] x 100 %

% of Total DR_{BASE} = **0.056%**

Therefore, the baseline percent of total dose rate due to Type A testing is 0.056%.

<u>Step 3 - Evaluate risk impact of extending Type A test interval from 10-to-15 years.</u>

The revised methodology in [REPORT02] suggests that a multiplier should be factored into the analysis to represent the change in probability of leakage. As stated in [REPORT03] and [NUREG01], relaxing the initial test interval from three ILRTs in a ten year period, to one ILRT in a ten year period increases the <u>average</u> time that a leak detectable only by an ILRT would go undetected from 18 months (3yrs / 2) to 60 months (10 yrs / 2). This is a factor of 3.333 (i.e. 60 / 18). The baseline dose associated with the ten-year interval was previously calculated using the percentage increase (10%), or 1.1 times the baseline dose. Using the 3.33 multiplier would yield a slightly higher ten-year dose. For a 15 year test interval, the <u>average</u> time that a leak detectable only by an ILRT would go undetected is 90 months (15 yrs / 2). Therefore, a factor of 5.0 (i.e. 90 / 18) should be applied.

<u>Risk Impact Related to 10-year Test Interval</u>

As previously stated, Type A tests impact only Class 1 and Class 3 sequences. In addition, the increased probability of not detecting excessive leakage has no impact on the frequency of occurrence for Class 1 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large liner opening remains the same, even though the probability of not detecting the liner opening increases). Thus, only the frequency of Class 3 sequences is impacted. Therefore, for Class 3 sequences, the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in probability of leakage of 3.33.

The increased leakage for the 10 year Class 3A and 3B frequencies are obtained by applying the 3.33 multiplier to the base values as shown below:

$FREQ_{Class3A10} = 7.91E\text{-}07 * 3.33 = 2.63E\text{-}06$ / year
$FREQ_{Class3B10} = 7.91E\text{-}08 * 3.33 = 2.63E\text{-}07$ / year

The frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3A and Class 3B in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$CLASS\text{-}1\text{-}FREQ_{10} = FREQ_{Class\text{-}1} - (FREQ_{Class3A} + FREQ_{Class3B})$

$CLASS\text{-}1\text{-}FREQ_{10} = 3.17E\text{-}06 - (2.63E\text{-}06 + 2.63E\text{-}07)$

$CLASS\text{-}1\text{-}FREQ_{10} = 2.77E\text{-}07$ / year

The results of these calculations are presented in Table 4 below.

Table 4: Mean Consequence Measures for 10-Year Test Interval - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Person-Rem (50-Miles)	Person-Rem/yr (50-Miles)
1	No Containment Failure	2.77E-07	5.98E+02	1.66E-04
2	Large Isolation Failures (Failure-to-close)	6.52E-10	2.38E+04	1.55E-05
3A	Small Isolation Failures (Type A test)	2.63E-06	5.98E+03	1.57E-02
3B	Large Isolation Failures (Type A test)	2.63E-07	2.09E+04	5.50E-03
4	Small isolation Failure-to-Seal (Type B test)	N/A	N/A	N/A
5	Small isolation Failure-to-Seal (Type C test)	N/A	N/A	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	N/A	N/A	N/A
7	Failure Induced by Phenomena (Early and Late Failures)	2.36E-05	1.00E+05	2.36
8	Bypass (SGTR)	2.49E-06	3.61E+06	8.99
CDF	All CET End States	**2.93E-05**	N/A	**11.37**

Based on the above values, the Type A 10-year test frequency percent of total dose rate for Class 3 is as follows:

$\% \text{ of Total } DR_{10} = [(CLASS3A_{10} + CLASS3B_{10}) / Total_{10}] \times 100$

<u>where</u>:

$CLASS3A_{10}$ = Class 3A person-rem/year = 1.57E-02 person-rem/year [Table 4]

$CLASS3B_{10}$ = Class 3B person-rem/year = 5.50E-03 person-rem/year [Table 4]

$Total_{10}$ = total person-rem year for 10-year interval = 11.37 person-rem/year [Table 4]

% of Total DR_{10} = [($CLASS3A_{10}$+ $CLASS3B_{10}$) / $Total_{10}$] x 100
% of Total DR_{10} = [(1.57E-02 + 5.50E-03) / 11.37] x 100
% of Total DR_{10} = 0.186%

Therefore, the total 10-year test frequency ILRT interval percent of total dose rate related to Type A testing is 0.186%.

The Δ% change in the 10 year ILRT Dose Rate from the baseline value is 0.186% - 0.056% = 0.130%.

The ten-year dose rate change (related to an ILRT) over the baseline case is as follows:

DR $Change_{10}$ = [$\sum^n$ (Class 1, 3A, 3B) $_{10}$ - $\sum^n$ (Class 1, 3A, 3B) $_{Base}$]

where:
$\sum^n$ (Class 1,3A,3B) $_{Base}$ = 1.38E-03 + 4.73E-03 + 1.65E-03 person-rem/year [Table 3]
$\sum^n$ (Class 1,3A,3B) $_{Base}$ = 7.76E-03 person-rem/year

$\sum^n$ (Class 1,3A,3B) $_{10}$ = 1.66E-04 + 1.57E-02 + 5.50E-03 person-rem/year [Table 4]
$\sum^n$ (Class 1,3A,3B) $_{10}$ = 2.14E-02 person-rem/year

DR $Change_{10}$ = [2.14E-02 - 7.76E-03] person-rem/year
DR $Change_{10}$ = 1.36E-02 person-rem/year

Therefore, the ten-year dose rate change from the baseline case is 1.36E-02 person-rem/year.

Risk Impact Related to 15-year Test Interval

The risk contribution for a 15 year interval is similar to the 10-year interval. The difference is in the increase in probability of leakage value. This increase in containment leakage is accounted for by using the multiplier 5.0 on the Class 3 frequencies.

The increased leakage for the 15 year Class 3A and 3B frequencies are obtained by applying the multiplier 5 to the base values as shown below:

$FREQ_{Class\ 3A15}$ = 7.91E-07 * 5.0 = 3.96E-06 / year
$FREQ_{Class\ 3B15}$ = 7.91E-08 * 5.0 = 3.96E-07 / year

The frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3A and Class 3B in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$CLASS\text{-}1\text{-}FREQ_{15} = FREQ_{Class\text{-}1} - (FREQ_{Class3A} + FREQ_{Class3B})$

$CLASS\text{-}1\text{-}FREQ_{15} = 3.17E\text{-}06 - (3.96E\text{-}06 + 3.96E\text{-}07)$

$CLASS\text{-}1\text{-}FREQ_{15} = -1.18E\text{-}06$ / year

Since the Class 1 frequency for the 15 year interval is less than zero (which is impossible), the Class 1 frequency will be set to zero and an over estimation of the total frequency will result. The results of this calculation are presented in Table 5 below.

Table 5: Mean Consequence Measures for 15-Year Test Interval - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Person-Rem (50-Miles)	Person-Rem/yr (50-Miles)
1	No Containment Failure	0.00E-00	5.98E+02	0.00E+00
2	Large Isolation Failures (Failure-to-close)	6.52E-10	2.38E+04	1.55E-05
3A	Small Isolation Failures (Type A test)	3.96E-06	5.98E+03	2.37E-02
3B	Large Isolation Failures (Type A test)	3.96E-07	2.09E+04	8.28E-03
4	Small isolation Failure-to-Seal (Type B test)	N/A	N/A	N/A
5	Small isolation Failure-to-Seal (Type C test)	N/A	N/A	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	N/A	N/A	N/A
7	Failure Induced by Phenomena (Early and Late Failures)	2.36E-05	1.00E+05	2.36E+00
8	Bypass (SGTR)	2.49E-06	3.61E+06	8.99E+00
Total	All CET End States	**3.04E-05***	N/A	**11.38**

***The CDF remains at 2.93E-05**

Based on the above values, the Type A 15 year test frequency percent of total dose rate (DR) for Class 3 is as follows:

$\% \text{ of Total } DR_{15} = [(CLASS3A_{15} + CLASS3B_{15}) / Total_{15}] \times 100$

<u>where</u>:

$CLASS3A_{15}$ = Class 3A person-rem/year = 2.37E-02 person-rem/year [Table 5]

$CLASS3B_{15}$ = Class 3B person-rem/year = 8.28E-03 person-rem/year [Table 5]

$Total_{15}$ = total person-rem/year for 15-year interval = 11.38 person-rem/year [Table 5]

$\% \text{ of Total } DR_{15} = [(2.37E\text{-}02 + 8.28E\text{-}03) / 11.38] \times 100$

$\% \text{ of Total } DR_{15} = 0.281\%$

Therefore, the total 15 year test frequency ILRT interval percent of total dose rate related to Type A testing is 0.281%.

The Δ% change in the 15 year ILRT DR from the baseline value is 0.281% - 0.056% = 0.225%.

The Δ% change in the total dose rate between the ten-to-fifteen year intervals related to Type A testing is:

$\Delta\%\ Change_{10\text{-}15} = \%\ of\ Total\ DR_{15} - \%\ of\ Total\ DR_{10} = 0.281\% - 0.186\% = 0.095\%$

The fifteen-year dose rate change (related to an ILRT) over the baseline case is as follows:

$DR\ Change_{15} = [\sum^{n}(Class\ 1, 3A, 3B)_{15} - \sum^{n}(Class\ 1, 3A, 3B)_{Base}]$

where:

$\sum^{n}(Class\ 1, 3A, 3B)_{Base}$ = 1.38E-03 + 4.73E-03 + 1.65E-03 person-rem/year [Table 3]

$\sum^{n}(Class\ 1, 3A, 3B)_{Base}$ = 7.76E-03 person-rem/year

$\sum^{n}(Class\ 1, 3A, 3B)_{15}$ = 0.00E+00 + 2.37E-02 + 8.28E-03 person-rem/year [Table 5]

$\sum^{n}(Class\ 1, 3A, 3B)_{15}$ = 3.20E-02 person-rem/year

$DR\ Change_{15}$ = [3.20E-02 - 7.76E-03] person-rem/year

$DR\ Change_{15}$ = 2.42E-02person-rem/year

Therefore, the fifteen-year dose rate change from the baseline case is 2.42E-02 person-rem/year.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF)

The one time extension of increasing the Type A test interval involves establishing the success criteria for a large release. This criterion is based on two prime issues:

1) The containment leak rate versus breach size, and

2) The impact on risk versus leak rate.

The containment leak size for the corresponding leak rate was calculated using the same methodology as in [CALC04]. The leak size and the corresponding leak rate data for Surry were used from [CALC08]. The effect of containment leak size on the containment leak rate is shown in Table 6. In addition, Oak Ridge National Laboratory (ORNL) [REPORT06] completed a study evaluating the impact of leak rates on public risk using information from WASH-1400 [REPORT07] as the basis for its risk sensitivity calculations (see Figure 1).

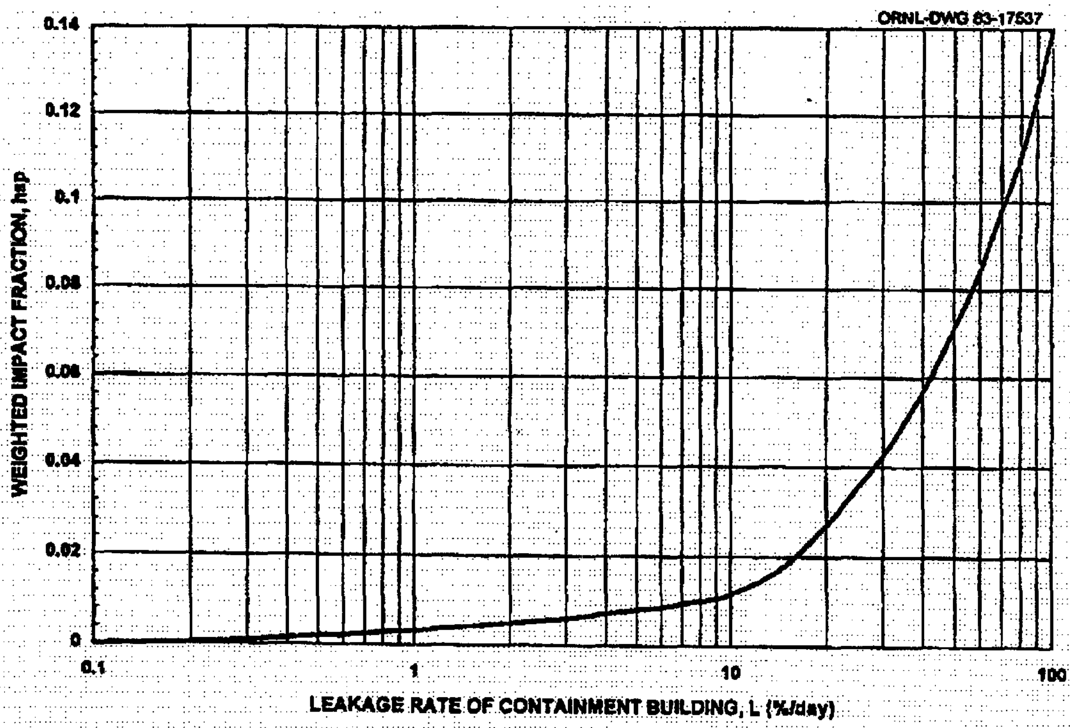
Based upon the information in Table 6 and ORNL, it is judged that small leaks resulting from a severe accident (that are deemed not to dominate public risk) can be defined as those that change risk by less than 5%. This definition would include leaks of less than 35%/day. Based on the Table 6 data, a 35%/day containment leak rate equates to a diameter leak of slightly smaller than 0.7 inches. It is to be noted that for Surry a containment leak with a diameter of 0.7 inches was calculated as opposed to 2.0 inches for Indian Point 3. This difference in containment leak diameter is due to the difference in containment size between Surry and Indian Point 3. Therefore, this study defines small leakage as containment leakage resulting from an opening of $0.012 in^2$ or less, large leakage as greater than $0.012 in^2$ and negligible leakage as $0.001 in^2$ or less.

Table 6: Evaluated Impact of Containment Leak Size on Containment Leak Rate

Containment Leak Size		**Approximate Containment Leak Rate at Design Pressure**
Diameter (inches)	**Area (in^2)**	**L_a (wt%/day)**
0.036	0.001	0.1 (acceptable by Tech Specs)
0.115	0.010	1.0 ($10L_a$)
0.126	0.012	3.5 ($35L_a$)
0.364	0.104	10.0
0.681	0.363	35.0
1.152	1.043	100.0
5.647	25.05	2400

The risk impact associated with extending the ILRT interval involves the potential of a core damage event, normally resulting in only a small radioactive release from containment, could in fact result in a large release due to failure to detect a pre-existing leak during the relaxation period. For this evaluation only Class 3 sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because for these sequences, the containment remains intact. Therefore, the containment leak rate is expected to be small (less than $1L_a$). A larger leak rate would imply an impaired containment, such as classes 2, 3, and 7.

Figure 1: Fractional Impact on Risk Associated with Containment Leak Rates [REPORT06]



Late releases are excluded regardless of the size of the leak because late releases are, by definition, not a LERF event. At the same time, sequences in the Surry IPE [REPORT09], which result in large releases (e.g., large isolation valve failures), are not impacted because a LERF will occur regardless of the presence of a pre-existing leak. Therefore, the frequency of Class 3B sequences (Table 4) is used as the LERF for Surry. This frequency, based on a ten-year test interval, is 2.63E-07/ yr.

Reg. Guide 1.174 [RG01] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Since the ILRT does not impact CDF, the relevant metric is LERF. Regulatory Guide 1.174 [RG01] states, when the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year (Region II). Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

Step 5 - Evaluate the Risk Impact in Terms of ΔLERF

The Δ LERF from Base to once-per-10 years (10 year metrics) is calculated to be the difference between the Class 3B frequencies in Tables 3 and 4.

Δ LERF = Class $3B_{10}$ – Class $3B_{Base}$
Δ LERF = 2.63E-07 – 7.91E-08 = 1.84E-07

The baseline total LERF for Surry has been calculated to be 6.62E-07 / yr in [NB01].
This Δ LERF increases the baseline LERF to 6.62E-07 + 1.84E-07 = 8.46E-07 / yr.

The Δ LERF from Base to once-per-15 years (15 year metrics) is calculated to be the difference between the Class 3B frequencies in Tables 3 and 5.

Δ LERF = Class $3B_{15}$ – Class $3B_{Base}$
Δ LERF = 3.96E-07 – 7.91E-08 = 3.17E-07

This Δ LERF increases the baseline LERF to 6.62E-07 + 3.17E-07 = 9.79E-07 / yr.

The Δ LERF from once-per-10 years to once-per-15 years (5 year metrics) is calculated to be the difference between the Class 3B frequencies in Tables 4 and 5.

Δ LERF = Class $3B_{15}$ – Class $3B_{10}$
Δ LERF = 3.96E-07 – 2.63E-07 = 1.33E-07

The guidance in [RG01] states that when the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be shown that the total LERF is less than 10^{-5} per reactor year. The total new LERF value for the 15 year change for SPS has been calculated to be 9.79E-07 / yr. Since guidance in [RG01] defines small changes in LERF, thus the magnitude in the difference between the 10 year and 15 year LERF value (1.33E-07) is in the range of 10^{-7}/yr to 10^{-6} /yr, increasing the ILRT interval to 15 years is considered acceptable.

Step 6 – Determine Impact on Conditional Containment Failure Probability

Another parameter that the NRC Guidance in [RG01] states can provide input into the decision making process is the consideration of 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 risk calculations performed in this analysis.

In this assessment, based on the NEI Interim Guidance [REPORT02], CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state (EPRI Category 1) and small pre-existing leakages (EPRI Category 3A). The conditional part of the definition is conditional given a severe accident (i.e., core damage). The CCFP percent for a given ILRT interval can be calculated using the following equation from [REPORT02]:

$CCFP_{\%}$ = [1-((Class1 Frequency + Class 3AFrequency) / Total CDF)] x 100%

For the Base interval, the values are obtained from Table 3:

$CCFP_{Base} = [1 - ((2.30E\text{-}06 + 7.91E\text{-}07) / 2.93E\text{-}05)] \times 100\%$
$CCFP_{Base} = 89.5\%$

For the 10-year interval, the values are obtained from Table 4:

$CCFP_{10} = [1 - ((2.77E\text{-}07 + 2.63E\text{-}06) / 2.93E\text{-}05)] \times 100\%$
$CCFP_{10} = 90.1\%$

For the 15-year interval, the values are obtained from Table 5:

$CCFP_{15} = [1 - ((0.00E\text{-}00 + 3.96E\text{-}06) / 2.93E\text{-}05)] \times 100\%$
$CCFP_{15} = 86.5\%$

The 5 year change (10 to 15 years) in the conditional containment failure probability is:

$\Delta CCFP_{\%} = CCFP_{15} - CCFP_{10} = -3.6\%$ (*see Note 1*)

The 10 year change in the conditional containment failure probability is:

$\Delta CCFP_{\%} = CCFP_{10} - CCFP_{Base} = 0.6\%$

The 15 year change in the conditional containment failure probability is:

$\Delta CCFP_{\%} = CCFP_{15} - CCFP_{Base} = -3.0\%$ (*see Note 1*)

This 15 year calculated change in $CCFP_{\%}$ is negative. Intuitively, this is not realistic and the expected 15 year change in the $CCFP_{\%}$ is negligible.

Note 1: *The negative $\Delta CCFP_{\%}$ is caused by the adjustment of the Class 1 frequency to preserve the total CDF. In reality, the factor of 5 increases of the Class 3 frequencies for the 15 year case cannot occur without an adjustment in other, more severe release categories.*

External Event Sensitivity Analysis

The SPS IPE [REPORT09] has limited discussion pertaining to external events, and it appears that external events would have the largest impact on the EPRI Class 7 event for the ILRT evaluation. However, in the Severe Accident Mitigation Alternatives analysis (SAMA) for the SPS license renewal [CALC02], a factor was used to account for the potential impact of external events. The benefits of each SAMA were multiplied by a factor of 2.0 to account for the external events. This factor could be applied to the CDF used here to calculate the EPRI Class 3A and 3B frequencies. Since Class 3B represents a LERF then this multiplier would have the following effect on the ILRT analysis.

Baseline Class 3B frequency = 7.91E-08 x 2.0 = 1.58E-07 /yr
15 year Class 3B frequency = 3.96E-07 x 2.0 = 7.92E-07 /yr

The external events change in LERF from the Baseline to the 15 year test interval is 6.34E-07 /yr (7.92E-07 – 1.58E-07). This compares to the internal events Baseline to 15 year change in LERF as 3.17E-07 /yr (3.96E-07 – 7.91E-08).

Thus it has been independently shown that with external events included, the change in LERF due to a 15 year ILRT interval still meets the screening criterion in [RG01]. Since guidance in [RG01] defines small changes in LERF in the range of 10^{-7}/yr, increasing the ILRT interval to 15 years is considered acceptable.

Linear Corrosion Analysis

The approach documented in the Calvert Cliffs Nuclear Power Plant submittal in [REPORT08] was used to determine the change in likelihood, due to extending the ILRT, of detecting liner corrosion. This likelihood was then used to determine the resulting change in risk. The following issues are addressed:

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

Assumptions

A. A half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 7, Step 1.)

B. The success data was limited to *5.5* years to reflect the years since September 1996 when 10 CFR *50.55a* started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date and there is no evidence that liner corrosion issues were identified (see Table 7, Step 1).

C. The 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 liner ages. Sensitivity studies are included that address the doubling of this rate every 10 years and every two years (see Table 7, Steps 2 and 3).

D. The likelihood of the containment atmosphere reaching the outside atmosphere given a liner flaw exists, is a function of the pressure inside the Containment. Even without the liner, the Containment is an excellent barrier. But as the pressure in Containment increases, cracks will form. If a crack occurs in the same region as a liner flaw, then the containment atmosphere can communicate to the outside atmosphere. At low pressures, this crack formation is extremely unlikely. Near the point of containment failure, crack formation is virtually guaranteed. Anchored points of 0.1% at 20 psia and 100% at 150 psia were selected. Intermediate failure likelihoods are determined through logarithmic interpolation. Sensitivity studies are included that decrease and increase the 20 psia anchor point by a factor of 10 (see Table 4 of [REPORT08] for sensitivity studies).

E. The likelihood of leakage escape (due to crack formation) in the basemat region is considered to be 10 times less likely than the containment cylinder and dome region (see Table 7, Step 4).

F. A 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection. (See Table 7, Step 5.) Sensitivity studies are included that evaluate total detection failure likelihoods of 5% (see Table 4 [REPORT08] for sensitivity studies).

G. All non-detectable containment over-pressurization failures are assumed to be large early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Table 7: Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome		Containment Basemat	
1	**Historical Liner Flaw Likelihood** Failure Data: Containment location specific. Success Data: Based on 70 steel-lined Containments and 9 years since the 10 CFR 50.55a requirement for periodic visual inspections of containment surfaces.	Events: 2 (Brunswick 2 and North Anna 2) 2/(70*5.5) = **5.2E-3**		Events: 0 Assume half a failure 0.5/(70*5.5) = **1.3E-3**	
2	**Aged Adjusted Liner Flaw Likelihood**	**Year**	**Flaw Likelihood**	**Year**	**Flaw Likelihood**
	During 15-year interval, assume	0	1.79E-03	0	4.47E-04
		1	2.05E-03	1	5.13E-04

Step	Description	Containment Cylinder and Dome		Containment Basemat	
	failure rate doubles every five years (i.e. a 14.9% increase per year). The average over the 5th through 10th year period was set to the historical failure rate of Step 1 (See Table-5 from [REPORT08] for an example). These assumptions are used to calculate the flaw likelihood for each year (for a 15 year period).	2 3 4 5 6 7 8 9 10 11 12 13 14 15	2.36E-03 2.71E-03 3.11E-03 3.57E-03 4.10E-03 4.71E-03 5.41E-03 6.22E-03 7.14E-03 8.21E-03 9.43E-03 1.08E-02 1.24E-02 1.43E-02	2 3 4 5 6 7 8 9 10 11 12 13 14 15	5.89E-04 6.77E-04 7.77E-04 8.93E-04 1.03E-03 1.18E-03 1.35E-03 1.55E-03 1.79E-03 2.05E-03 2.36E-03 2.71E-03 3.11E-03 3.57E-03
3	**Increase in Flaw Likelihood Between 3, 10, and 15 years** This cumulative probability uses the age adjusted liner flaw likelihood of Step 2 (see Tables 5 and 6 in [REPORT08]). For example, the 7.12E-03 (at 3 years) cumulative flaw likelihood is the sum of the year 1, year 2, and year 3 likelihoods of step 2.	**0.71% (1 to 3 years)** **4.14% (1 to 10 years)** **9.65% (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 ΔLERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the desired presentation of the results.)		**0.18% (1 to 3 years)** **1.03% (1 to 10 years)** **2.41% (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 ΔLERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the desired presentation of the results.)	
4	**Likelihood of Breach in Containment given Liner Flaw** The upper end pressure is consistent with the Calvert Cliffs Probabilistic Risk Assessment (PRA) Level 2 analysis. 0.1% is assumed for the lower end. Intermediate failure likelihoods are determined through logarithmic interpolation. The basemat is assumed to be 1/10 of the cylinder/dome analysis.	Pressure (psia) 20 **64.7 (ILRT)** 100 120 150	Likelihood of Breach 0.1% **1.1%** 7.02% 20.3% 100%	Pressure (psia) 20 **64.7 (ILRT)** 100 120 150	Likelihood of Breach 0.01% **0.11%** 0.7% 2.0% 10%

Step	Description	Containment Cylinder and Dome	Containment Basemat
	The same value will be used for SPS as was used for CCNP, since the containment design is somewhat similar. The design pressure of SPS is 45 psig versus 50 psig for CCNPP.		
5	**Visual Inspection Detection Failure Likelihood**	**10%** 5% failure to identity visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3*4*5)	**0.0106%** 9.65%*1.1%*10%	**0.0027%** 2.41%*0.11%*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.

Total Likelihood of Non-Detected Containment Leakage = 0.0106% + 0.0027% = 0.0133%

The non-large early release frequency (LERF) containment over-pressurization failure for SPS is estimated at 2.44E-05 per year. This is based on the total CDF minus the Class 1, 3B and 8 frequencies from Table 1 (2.44E-05 = 2.93E-05 – (2.30E-06 + 7.91E-08 + 2.49E-06)). The total CDF for SPS is 2.93E-05. If all non-detectable containment leakage events are considered to be LERF, then the increase in LERF associated with the liner corrosion issue is:

Increase in LERF (ILRT 3 to 15 years) = 0.000133 * 2.44E-05 = 3.25E-09 per year

Thus it has been independently shown that the increase in LERF due to a liner corrosion failure is 3.25E-09 per year which meets the screening criterion of less than 10^{-7} in [RG01].

5.0 RESULTS AND CONCLUSIONS

The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once-per-fifteen-years is 1.33E-07 / yr. Therefore, the risk impact when compared to other severe accident risks is small. The results for the Baseline, 10 year and 15 year ILRT evaluation are summarized in Table 8 below.

Table 8: Summary of Results

EPRI Class	Base Freq / Rx-yr	Base (3/10) Freq / Rx-yr	Base Dose Person-Rem	Base Dose Rate Person-Rem / yr	10yr Freq / Rx-yr	10yr Dose Rate Person-Rem / yr	10yr Metrics	15yr Freq / Rx-yr	15yr Dose Rate Person-Rem/yr	15yr Metrics
1	3.17E-06	2.30E-06	5.98E+02	1.38E-03	2.77E-07	1.66E-04		0.00E-00	0.00E+00	
2	6.52E-10	6.52E-10	2.38E+04	1.55E-05	6.52E-10	1.55E-05		6.52E-10	1.55E-05	
3A	N/A	7.91E-07	5.98E+03	4.73E-03	2.63E-06	1.57E-02		3.96E-06	2.37E-02	
3B	N/A	7.91E-08	2.09E+04	1.65E-03	2.63E-07	5.50E-03		3.96E-07	8.28E-03	
4	N/A	N/A	N/A	N/A	N/A	N/A		N/A	N/A	
5	N/A	N/A	N/A	N/A	N/A	N/A		N/A	N/A	
6	N/A	N/A	N/A	N/A	N/A	N/A		N/A	N/A	
7	2.36E-05	2.36E-05	1.00E+05	2.36	2.36E-05	2.36		2.36E-05	2.36E+00	
8	2.49E-06	2.49E-06	3.61E+06	8.99	2.49E-06	8.99		2.49E-06	8.99E+00	
TOTAL	**2.93E-05**	**2.93E-05**	**N/A**	**11.36**	**2.93E-05**	**11.37**		**3.04E-05**	**11.38**	
	(CDF)	(CDF)			(CDF)			(CDF)		
ΔDR Change from Base							1.36E-02			2.42E-02
ILRT DR % of total dose				0.056%			0.186%			0.281%
Δ% Change in ILRT DR from Base							0.130%			.0225%
LERF				6.62E-07			8.46E-07			9.79E-07
Δ LERF From base							1.84E-07			3.17E-07
CCFP, %				89.5%			90.1%			87.0%
Δ CCFP, %							0.60%			-2.50%
CDF	**2.93E-05**									

6.0 REFERENCES

[CALC01] SM-1241 Revision 0, "MACCS2 model for Surry Level 3 Application", 2-28-2000

[CALC02] SM-1256 Revision 1, "Surry Severe Accident Mitigation Alternative (SAMA)", 09-04-2000.

[CALC03] IP3-CALC-VC-03357 Revision 0, "Indian Point 3 Risk Impact Assessment of Extending Containment Type A Test Interval", 1-4-2001

[CALC04] SM-1237 Revision 0, "Surry and North Anna Containment Isolation Modeling",4-20-2000

[CALC05] SM-1237 Revision 0, Addendum A "Surry and North Anna Containment Isolation Modeling", 4-24-2001

[CALC06] SM-1325 Revision 0, Addendum A "Risk Impact Assessment of Extending Containment Type A Test Interval at North Anna Power Station", 10-10-2001

[CALC07] PRA06NQA-04178S3 Revision 2, "Risk Impact Assessment of Extending Containment Type A Test Interval at Millstone Unit 3", 11-21-2003

[CALC08] SM-1321 Revision 0, "Risk Impact Assessment of Extending Containment Type A Test Interval at Surry Power Station", 8-14-2001

[CALC09] C467060036-6915 Revision 0, "LaSalle ILRT Interval Extension Risk Assessment", March 2006

[CALC10] Florida Power Calculation, F-01-0001, Revision 2, "Evaluation of Risk Significance of ILRT Extension", 6-19-01

[CALC11] SM-1345 Revision 0, "Revised LERF Fractions Using Updated Surry Level 2 Model", November 2003

[NB01] PRA Model Notebook QU.2 Rev.3, "Model Quantification Results," Surry Power Station Units 1&2, June 2006

[NUREG01] NUREG-1493, "Performance-Based Containment Leak-Test Program", July 1995

[NUREG02] NUREG-1335, "Individual Plant Examination: Submittal Guidance," August 1989.

[PROCDR01] Procedure, Engineering Periodic Test 2-NPT-CT-101, "Reactor Containment Building Integrated Leak Rate Test (Type A Containment Testing)", Revision 5

[REPORT01] NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J", July 26, 1995, Revision 0

[REPORT02] 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, November 2001

[REPORT03] EPRI TR-104285, "Risk Assessment of Revised Containment Leak Rate Testing Intervals", August 1994

[REPORT04] EPRI TR-109325, "Risk Impact Assessment of Extended Leak Rate Testing Intervals" December 2003

[REPORT05] Z. T. Mendoza, et al., "Generic Framework for Individual Plant Examination (IPE) Backend (Level 2) Analysis, Volume 1 - Main Report and Volume 3 - BWR Implementation Guidelines," prepared by SAIC International, Inc., Electrical Power Research Institute, NSAC-159, EPRI PR3114-29, 1991

[REPORT06] Burns, T.J., "Impact of Containment Building Leakage on LWR Accident Risk", Oak Ridge National Laboratory, NUREG/CR-3539, April 1984

[REPORT07] United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975

[REPORT08] Calvert Cliffs Nuclear Power Plant, Letter from Mr. Charles H. Cruse to NRC Document Control Desk, "Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leak Rate Test Extension", dated March 27, 2002

[REPORT09] Probabilistic Risk Assessment Final Report "Surry Power Station Units 1 and 2 Individual Plant Examination", August 1991

[RG01] Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-informed Decisions On Plant-Specific Changes to the Licensing Basis" November 2002, Revision 1

ATTACHMENT A: SPS FREQUENCY AND DOSE DATA

Table A-1: SPS Frequency and Dose Data

Release Category	Frequency* Per year	Person-Rem**	EPRI Class	Description
1	0.00E+00	-------------	1	No CF
2	2.55E-06	5.98E+02	1	No CF
3	0.00E+00	5.15E+06 [a]	7	Early CF
4	1.41E-07	2.50E+04 [a]	7	Late CF
5	9.16E-09	8.23E+05 [a]	7	Late CF
6	7.79E-09	2.50E+04 [a]	7	Late CF
7	8.10E-07	8.23E+05 [a]	7	Late CF
8	4.43E-07	2.89E+05 [a]	7	Late CF
9	1.35E-05	7.10E+04 [a]	7	Late CF
10	8.45E-06	7.10E+04 [a]	7	Late CF
11	2.89E-07	2.50E+04	7	Melthru
12	0.00E+00	5.98E+02 [a]	2	No Cont. Iso
13	6.52E-10	4.71E+05 [a]	2	No Cont. Iso
14	6.23E-07	-------------	1	Debris Cool IV
15	0.00E+00	1.19E+04 [b]	2	Debris Cool IV
16	0.00E+00	1.19E+04 [b]	2	Debris Cool IV
17	1.37E-06	2.75E+06 [a]	8	Event V (attenuation)
18	2.42E-07	6.81E+06 [a]	8	Event V (no attenuation)
19	5.33E-07	5.07E+06 [a]	8	SGTR
20	3.42E-07	2.54E+06 [c]	8	SGTR (non-LERF)
CDF Freq	2.93E-05			

* Frequency data taken from [CALC11].
**Person-Rem data taken from [CALC01].
[a] Recommended Alternate values were used consistent with the IPE, SAMA analysis, and [CALC01].
[b] Used IPE STC 21 for STC 15 and 16 based on review of MAAP runs.
[c] Dose for STC 20 was assumed to be half that of STC 19 because it is a non-LERF event.

TABLE A-2: Surry Total EPRI Class Frequency and Dose Data

	Frequency	Dose
Class 1	3.17E-06	5.98E+02
Class 2	6.52E-10	2.38E+04
Class 7	2.36E-05	7.30E+06
Class 8	2.49E-06	1.72E+08

ATTACHMENT B: JUSTIFICATION OF VOLUME CHANGE

Revision 0

This is the original revision.

Revision 1

Editorial revision for enhancement and clarity prior to NRC LAR submittal.

Revision 2

Revised to clarify why the Conditional Containment Failure Probability was negative.

ATTACHMENT C: REVIEWER COMMENTS/RESOLUTIONS

Comment Number	Section/ Page	Review Comment	Response to Review Comment
1	All	Minor editorial comments	Corrected
2	All	Editorial revisions for enhancement and clarity	Done
3	14 & 19	Clarification of negative CDF and CCFP calculations	Corrected

Attachment 3

Marked-up Technical Specifications Page

Surry Power Station
Unit 2
Virginia Electric and Power Company
(Dominion)

4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage testing.

Objective

To assure that leakage of the primary reactor containment and associated systems is held within allowable leakage rate limits; and to assure that periodic surveillance is performed to assure proper maintenance and leak repair during the service life of the containment.

Specification

A. Periodic and post-operational integrated leakage rate tests of the containment shall be performed in accordance with the requirements of 10 CFR 50, Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors."

B. Containment Leakage Rate Testing Requirements

1. The containment and containment penetrations leakage rate shall be demonstrated by performing leakage rate testing as required by 10 CFR 50 Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September, 1995 as modified by the following exception:

   NEI 94-01-1995, Section 9.2.3: The first Unit 2~~1~~ Type A test performed after the October 26, 2000~~April 23, 1992~~ Type A test shall be performed no later than October 26, 2015~~April 22, 2007~~.

2. Leakage rate acceptance criteria are as follows:

   a. An overall integrated leakage rate of less than or equal to La, 0.1 percent by weight of containment air per 24 hours, at calculated peak pressure (Pa).

   b. A combined leakage rate of less than or equal to 0.60 La for all penetrations and valves subject to Type B and C testing when pressurized to Pa.

   Prior to entering an operating condition where containment integrity is required the as-left Type A leakage rate shall not exceed 0.75 La and the combined leakage rate of all penetrations subject to Type B and C testing shall not exceed 0.6 La.

3. The provisions of Specification 4.0.2 are not applicable.

Basis

The leak tightness testing of all liner welds was performed during construction by welding a structural steel test channel over each weld seam and performing soap bubble and halogen leak tests.

Attachment 4

Proposed Technical Specifications Page

Surry Power Station
Unit 2
Virginia Electric and Power Company
(Dominion)

4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage testing.

Objective

To assure that leakage of the primary reactor containment and associated systems is held within allowable leakage rate limits; and to assure that periodic surveillance is performed to assure proper maintenance and leak repair during the service life of the containment.

Specification

A. Periodic and post-operational integrated leakage rate tests of the containment shall be performed in accordance with the requirements of 10 CFR 50, Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors."

B. Containment Leakage Rate Testing Requirements

1. The containment and containment penetrations leakage rate shall be demonstrated by performing leakage rate testing as required by 10 CFR 50 Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September, 1995 as modified by the following exception:

    NEI 94-01-1995, Section 9.2.3: The first Unit 2 Type A test performed after the October 26, 2000 Type A test shall be performed no later than October 26, 2015.

2. Leakage rate acceptance criteria are as follows:

    a. An overall integrated leakage rate of less than or equal to La, 0.1 percent by weight of containment air per 24 hours, at calculated peak pressure (Pa).

    b. A combined leakage rate of less than or equal to 0.60 La for all penetrations and valves subject to Type B and C testing when pressurized to Pa.

    Prior to entering an operating condition where containment integrity is required the as-left Type A leakage rate shall not exceed 0.75 La and the combined leakage rate of all penetrations subject to Type B and C testing shall not exceed 0.6 La.

3. The provisions of Specification 4.0.2 are not applicable.

Basis

The leak tightness testing of all liner welds was performed during construction by welding a structural steel test channel over each weld seam and performing soap bubble and halogen leak tests.