VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 5, 2007

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

Serial No. 07-0769
NL&OS/ETS R0
Docket No. 50-339
License No. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA 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 NPF-7 for North Anna Power Station Unit 2. The proposed change will permit a one-time five-year exception to the ten (10) 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 9, 2014.

Attachments 1 and 2 provide an evaluation and the risk assessment for the proposed change. 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 Station Nuclear Safety and Operating Committee.

To permit effective outage planning, Dominion requests approval of the proposed Technical Specification change by June 30, 2008. Upon issuance, the amendment will be implemented within 30 days.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the appropriate designated officials of Virginia.

If you have any questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Very truly yours,

Gerald T. Bischof
Vice President – Nuclear Engineering

Attachments

1. Evaluation of Proposed License Amendment
2. Probabilistic Risk Assessment
3. Marked-up Technical Specifications Change
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 Gerald T. Bischof, who is Vice President – Nuclear Engineering, 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 5TH day of December, 2007.

My Commission Expires: May 31, 2010.

Vicki L. Hull
Notary Public

(SEAL)

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My Commission Expires May 31, 2010

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Attachment 1

Evaluation of Proposed License Amendment

North Anna 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 Containment Leakage Rate Testing Program to permit a five year extension of the North Anna Unit 2 integrated leak rate test. The proposed change will permit a one-time five-year exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by Regulatory Guide (RG) 1.163 (Reference 3) and NEI 94-01 (Reference 1). The one-time exception is to the requirement of NEI 94-01 to perform Type A test at a frequency of up to ten years for North Anna Unit 2.

2.0 PROPOSED CHANGE

This application for amendment to the North Anna Unit 2 Technical Specifications proposes to add an exception to the Containment Leakage Rate Testing Program. Specifically, this revision takes a one-time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by RG 1.163 and NEI 94-01. The exception allows the performance of a Type A test within fifteen years from the last Type A test, which was performed on October 9, 1999. 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 associate with the test frequency. The specific change to Unit 2 TS 5.5.15, “Containment Leakage Rate Testing Program” is as follows:

- Revise the current Unit 1 exception to NEI 94-01 to a Unit 2 exception to NEI 94-01. The exception reads as follows:

  "NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A Test performed after the October 9, 1999 Type A test shall be performed no later than October 9, 2014."

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 considers 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 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.”

The North Anna Power Station Unit 2 current ten-year Type A test interval ends on October 9, 2009. In order to meet the interval requirements of NEI 94-01, this test must be performed during refueling outage, N2R-19, scheduled to commence in September of 2008. By granting the proposed one-time exception, North Anna 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 $1.4 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." Amendment 177 (Reference 2) was issued to North Anna Power Station Unit 2 to permit implementation of 10 CFR 50, Appendix J, Option B. Amendment 177 modified Technical Specification Section 4.6.1.1 to require testing in accordance with the Containment Leakage Rate Testing Program and RG 1.163 (Reference 3), respectively. This requirement was subsequently incorporated into Section 5.5.15, "Containment Leakage Rate Testing Program," during the ITS conversion in 2002. 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.

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 consistent with NEI 94-01. Based on the October 1990 and October 1999 Type A tests, the current interval for North Anna Unit 2 is once every ten years.

4.2 North Anna Integrated Leak Rate 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 North Anna 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 North Anna Unit 2.

Test Date	As-Found Leakage			Acceptance Limit (*)
April 1989	Measured Leakage		0.24 of L_a	
	Upper confidence limit (UCL) Margin		0.03 of L_a	
	Type C Penalty		0.03 of L_a	
	Non-vented Penalty		0.00 of L_a	
	TOTAL		**0.30** of L_a	1.0L_a
October 1990	Measured Leakage	--	0.20 of L_a	
	UCL Margin	--	0.02 of L_a	
	Type C Penalty	--	0.03 of L_a	
	Non-vented Penalties	--	0.00 of L_a	
	TOTAL		**0.25** of L_a	1.0L_a
October 1999	Measured Leakage	--	0.450 of L_a	
	UCL Margin	--	0.004 of L_a	
	Type C Penalty	--	0.090 of L_a	
	Non-vented Penalties	--	0.040 of L_a	
	TOTAL		**0.614** 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 reactor containment structure 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 structure is constructed inside an open cut excavation in rock. The structure is rock-supported. The base of the foundation mat is located approximately 67 feet below finished ground grade. The containment structure has an inside diameter of 126 ft. 0 in. The bend line of the dome is 127 ft. 7 in. above the top of the foundation mat. The inside radius of the dome is 63 ft. 0 in.

The interior vertical height is 190 ft. 7 in. measured from the top of the foundation mat to the center of the dome. The cylindrical wall is 4 ft. 6 in. thick, the dome is 2 ft. 6 in. thick, and the base mat is 10 ft. 0 in. thick. The steel liner for the wall is 3/8 inch thick. The steel liner for the mat consists of a 0.25-inch plate except: in the incore instrumentation area, where an exposed 0.75-inch plate is used; and the inside recirculation spray pump sumps, where an exposed 0.5-inch plate is used. The steel liner for the dome is 0.5 inch thick. A waterproof membrane was placed below the containment structural mat and carried up the containment wall to above ground-water level. Attached to and entirely enveloping the structure below grade, the membrane protects concrete reinforcing from ground-water corrosion, and the steel liner from external hydrostatic pressure.

Access to the containment structure is provided by a 7 ft. 0 in. inside diameter (ID) personnel hatch and a 14 ft. 6 in. ID equipment hatch. Other smaller containment structure penetrations include hot and cold pipes, main steam and feedwater pipes, the fuel transfer tube, and electrical conductors. The reinforced-concrete structure is designed to withstand all loadings and stresses anticipated during the operation and life of the plant. The steel liner is attached to and supported by the concrete. The liner functions primarily as a gas tight membrane, and transmits loads to the concrete. During construction, the steel liner served as the inside form for the concrete wall and dome. The containment structure does not require the participation of the liner as a structural component. No credit is taken for the presence of the steel liner in the design of the containment structure to resist seismic forces or other design loads.

The steel wall and dome liner are protected from potential interior missiles by interior concrete shield walls. The base mat liner is protected by a 21-inch to 30-inch thick concrete cover, except in the incore instrumentation area, the inside recirculation spray pump sumps, the containment drainage sumps, the low end of the containment sump trench, where the slope results in a minimum of approximately 12 inches of concrete cover, and the bottom of the containment sump.

The safety design basis for the containment is that the containment must withstand the pressure and temperatures of the limiting design basis accident (DBA) without exceeding the design leakage rate.

Containment air partial pressure is an initial condition used in the containment DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered relative to containment pressure are the loss of coolant accident (LOCA) and steam line break (SLB). The LOCA and SLB are assumed not to occur simultaneously or consecutively. The containment analysis for the DBA shows that the maximum peak containment pressure results from the limiting design basis SLB. The maximum design internal pressure for the containment is 45.0 psig. The LOCA and SLB analyses establish the limits for the containment air partial pressure operating range. This resulted in a Unit 2 maximum peak containment internal pressure of 42.7 psig for a LOCA, which is less than the maximum design internal pressure for the containment. The Unit 2 SLB analysis resulted in a maximum peak containment internal pressure of 43.0 psig, which is also less than the maximum design internal pressure for the containment.

The containment was also designed for an external pressure load of 9.2 psid (i.e., a design minimum pressure of 5.5 psi). The inadvertent actuation of the Quench Spray (QS) System was analyzed to determine the reduction in containment pressure remains within the containment minimum design pressure.

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 North Anna TS 5.5.15.d). 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, North Anna Unit 2 is maintained at a subatmospheric condition (see TS 3.6.4). Containment air partial pressure is maintained with an operating range (10.3 psia to 12.3 psia) based on service water temperature to ensure the containment design pressure is not exceeded during a design basis accident. Instrumentation constantly monitors containment pressure. If pressure rises, an alarm annunciates conditions approaching the limits allowed by 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 tests

North Anna Unit 2 has completed the requirements of their first ten-year containment inservice inspection program. Concrete containment examinations (IWL) 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 examinations. These examinations on the concrete exterior were conducted by the Responsible Engineer using the visual method (VT-3C and VT-C). 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 no later than one year from the specified date. North Anna 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 performed to the 1992 Edition with the 1992 Addenda of ASME Section XI with the exception of 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 North Anna Unit 2 is March 20, 1998 to October 11, 2008.

Examinations performed for North Anna Unit 2 include the General Visual of all accessible areas, a visual (VT-1) examination of pressure retaining bolting, and augmented visual (VT-1) and ultrasonic thickness examinations on areas associated with the identified liner wall damage caused by wood left in the concrete at construction. 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, E-C, and E-G of the applicable Code. No other examination categories apply.

The second ten-year interval IWE examination requirements will use the 2001 Edition through the 2003 Addenda of ASME Section XI as modified by the 10 CFR 50.55a (b) limitations for both units. At this time no augmented Category E-C examinations are planned. Augmented exams on the North Anna Unit 2 containment liner have not identified wall thickness changes following the containment liner repair activity after several examinations (following the removal of the wood from behind the liner in fall 1999). 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 North Anna 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 North Anna 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 extension requested for North Anna Unit 2 only applies to the 10 CFR 50, Appendix J, Type A integrated leak rate 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.

The second ten-year interval IWE program for North Anna 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 the North Anna units, 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.

The 2002 IWL containment inspections of the North Anna Unit 2 containment structure identified embedded material in the containment dome areas. There were several pieces of wood embedded in the surface of the concrete. With the exception of three pieces of wood, the wood pieces were small and when removed, concrete repair was not required. In the other three cases, the wood extended into the concrete. These pieces were removed and although no structural issues were identified, the voids were grouted to prevent moisture intrusion.

The embedded material, as described above, was inadvertently left in the containment structure during original plant construction. The slight depression of the wood 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 identified in the 2002 inspection. The assessment concluded that the containment structure remained fully capable of meeting the functional design requirements as described in Technical Specification 3.6.1 and UFSAR Section 3.8.2. This assessment assumed that the wood extended from the concrete surface through the concrete placement. The engineering assessment of the inspection findings concluded that:

- the leak-tight integrity of the liner has not been jeopardized,
- no degradation of reinforcing steel was identified,
- 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.

The second 5-year IWL containment inspections of the North Anna Unit 2 containment structure were completed in August of 2007. Similar to the 2002 inspections, this inspection identified several pieces of embedded material in the assessable portions of containment. An engineering assessment of the containment structural integrity based on the identified defects, taken together or individually, do 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 2002 refueling outage North Anna Unit 2 replaced the reactor vessel head. To complete the replacement required an opening in the containment larger than the equipment hatch. Therefore, an opening ranging from a size of approximately 13 ft.0" high by 17 ft. 0 in. wide in the steel liner plate to a size of approximately 20 ft. 0 in. high by 25 ft. 0 in. wide at the outside face of the concrete was made. 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. The nondestructive examination of the containment liner was in accordance with Safety Guide 19, "Nondestructive Examination of Primary Containment Liners," with the following changes: 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 criteria 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 in accordance with NRC approved relief request RR-IWE9 (Reference 8).

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 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.

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 9) provides guidance for determining the risk impact of permanent plant-specific changes to the licensing basis. Since the Type A test interval does not impact core damage frequency (CDF) the relevant criterion is the Large Early Release Frequency (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 1.58E-7/yr, based on internal events. RG 1.174 states that when the calculated (permanent) increase in LERF is in the range of 1E-7/yr to 1E-6/yr, the proposed increase is "small" and the application will be considered when the baseline LERF is less than 1E-5/yr. Since the baseline LERF for North Anna is 1.20E-6/yr, the proposed one-time change is bounded by this threshold.

- 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 0.025%. 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.' For the current ten-year Type A test interval, the contribution of sequences involving containment failure for the ten-year interval is 50.7%. For the proposed fifteen-year interval, the contribution of sequences involving containment failure increased to 51.1%. Therefore, the $\Delta CCFP_{10\text{-}15}$ is found to be 0.40%. This represents a small change in the North Anna Unit 2 containment defense-in-depth.

The risk assessment calculation performed for the five-year Type A test extension for Unit 2 from ten years to fifteen years is included as 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.

North Anna 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 three 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 North Anna 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 North Anna 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. 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 North Anna 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 very small changes in risk as resulting in increases of CDF below 1E-6/yr and increases in LERF below 1E-7/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 1.58E-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 1.20E-6yr, 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 North Anna, 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 IV 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 very small changes in risk as resulting in increases of CDF below 1E-6/yr and increases in LERF below 1E-7/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 a once-per-ten-years to a once-per-fifteen-years is 1.58E-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 1.20E-6/yr, then the change is considered acceptable.

The Station Nuclear Safety and Operating Committee (SNSOC) has reviewed this proposed change to the Technical Specifications and has concluded that it does not involve a significant hazards consideration and will not endanger the health and safety of the public.

The proposed change has been reviewed and it has been determined that the change qualifies for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change.

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 North Anna Issuing Technical Specification Amendment 177, dated February 9, 1996 to implement 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 ANSI/ANS - 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. NEI 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 North Anna approving RR-IWE9, "North Anna Power Station, Unit 2 - ASME Section XI, Inservice Inspection Program, Relief Request (RR) RR-IWE9 for Containment Testing," dated January 14, 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 A Extension for North Anna Unit 2

North Anna Power Station Unit 2
Virginia Electric and Power Company
(Dominion)

North Anna Power Station
Probabilistic Risk Assessment Notebook

Part V PRA Risk Analysis

Volume RA.LI.3

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

Revision No. 1
Effective Date: November 2007

Purpose: To provide a risk impact assessment on extending the Integrated Leak Rate Test (ILRT) interval for North Anna Unit 2 from 10 years to 15 years.		
Conclusion: The increase in Large, Early Release Frequency (LERF) resulting from a change in the Type A ILRT test interval from once-per-ten-years to once-per-fifteen-years is 1.58E-07 / yr. Therefore, the risk impact when compared to other severe accident risks and the acceptance criteria in Regulatory Guide 1.174 is small.		
Prepared By:	**Signature**	**Date**
Reviewed By:	**Signature**	**Date**
Approved By:	**Signature**	**Date**

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 North Anna 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 North Anna Nuclear Power Station (NAPS) selected the requirements under Option B as its testing program [PROCDR01].

The surveillance testing requirement 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 North Anna Unit 2 current 10-year Type A test interval ends in October 2009. The proposed amendment to the 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 1999.

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 North Anna Individual Plant Examination (IPE) [REPORT09], the revised model [CALC02] and [CALC04], and previous calculations [CALC06], [CALC07], and [CALC09] were used for this risk assessment calculation.

3.0 ANALYSIS

3.1 Inputs

This calculation will use North Anna fifty 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 IPE [REPORT09] and revised data in [CALC02]. Source term category is defined here as a grouping of like releases of 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 North Anna 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 go 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.

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 NAPS 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 both 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 manway leakage (EPRI TR-104285 Class 3 sequences). A Type B test measures component leakage across pressure retaining boundaries (e.g. gaskets, expansion bellows and air locks). A Type C test measures component leakage rates across the 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) were not accounted for in this evaluation (EPRI TR-104285 Class 6 sequences).

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

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 the EPRI Report [REPORT03]. See Table A-1 of Attachment A.

Step 2 - Develop baseline plant specific person-radiation dose in rem (population dose) per reactor year for each of the eight accident classes evaluated in EPRI TR-104285 [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 NAPS 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 the plant features that influence the phenomena.

As previously described, the 5 year 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. 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 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. NAPS 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 been isolated but then 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 frequency and will be calculated first.

Class 3 Sequences: This group consists of all core damage accident progressions collected (binned) 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] were used. 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 and Table 6 in Step 4 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 c^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 c^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 = 3.50E-05 / year [Table A-1, Attachment A]

CLASS-3A-Base-Frequency = 0.027 * 3.50E-05 / year = 9.45E-07 / year

CLASS-3B-Base-Frequency = 0.0027 * 3.50E-05 / year = 9.45E-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 1.76×10^{-5} / year (see Attachment A, Table A-1). For this analysis the associated maximum containment leakage for this group is $1L_a$. The NAPS 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:

CLASS-1-FREQ = $FREQ_{Class\text{-}1}$ - ($FREQ_{Class3A}$ + $FREQ_{Class3B}$)
CLASS-1-FREQ = 1.76E-05 - (9.45E-07 + 9.45E-08)

CLASS-1-Base-Frequency = 1.66E-05 / 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 = 6.45E-06 / 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 = 4.99E-06 / 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, Attachment A as Class 8.

CLASS-8-FREQUENCY = 5.89E-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	1.66E-05
2	Large Containment Isolation Failures (Failure-to-close)	6.45E-06
3A	Small Isolation Failures (Type A test)	9.45E-07
3B	Large Isolation Failures (Type A test)	9.45E-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)	4.99E-06
8	Containment Bypassed (SGTR & V-Sequence)	5.89E-06
CDF	Core Damage All CET End states	**3.50E-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 North Anna 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, Attachment A. 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, Attachment A represents the sum of the doses for the Event-V and SGTR sequences. The baseline dose results are calculated below and are summarized in Table 2.

Class 1 = 4.24E+02 person-rem
Class 2 = 1.85E+06 person-rem
Class 3A = 4.24E+02 * 10 = 4.24E+03person-rem
Class 3B = 4.24E+02 * 35 = 1.48E+04person-rem
Class 4 = Not analyzed
Class 5 = Not analyzed
Class 6 = Not analyzed
Class 7 = $\sum^n$ (Freq x Dose) / $\sum^n$ Freq = 5.68 x 10^4 person-rem
Class 8 = $\sum^n$ (Freq x Dose) / $\sum^n$ Freq = 4.26 x 10^6 person-rem

Table 2: Person-Rem Measures - Given Accident Class

Class	Description	Person-Rem (50-Miles)
1	No Containment Failure	4.24E+02
2	Large Containment Isolation Failures (Failure-to-close)	1.85E+06
3A	Small Isolation Failures (Type A test)	4.24E+03
3B	Large Isolation Failures (Type A test)	1.48E+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)	5.68E+04
8	Containment Bypassed (SGTR & V-Sequence)	4.26E+06

The above dose results when combined with the frequency results presented in Table 1 yields the NAPS 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	1.66E-05	4.24E+02	7.04E-03
2	Large Isolation Failures (Failure-to-close)	6.45E-06	1.85E+06	11.93
3A	Small Isolation Failures (Type A test)	9.45E-07	4.24E+03	4.01E-03
3B	Large Isolation Failures (Type A test)	9.45E-08	1.48E+04	1.40E-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)	4.99E-06	5.68E+04	0.28
8	Containment Bypassed (SGTR & V-Sequence)	5.89E-06	4.26E+06	25.09
CDF	All CET End States	**3.50E-05**	N/A	**37.31**

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 = 4.01E-03person-rem/year [Table 3]

$CLASS3B_{BASE}$ = class 3B person-rem/year = 1.40E-03person-rem/year [Table 3]

$Total_{BASE}$ = total person-rem/year for baseline interval = 37.31person-rem/year [Table 3]

% of Total DR_{BASE} = [($CLASS3A_{BASE}$ + $CLASS3B_{BASE}$) / $Total_{BASE}$] x 100%
% of Total DR_{BASE} = [(4.01E-03 + 1.40E-03) / 37.31] x 100 %

% of Total DR_{BASE} = **0.015%**

Therefore, the baseline percent of total dose rate related to Type A testing is 0.015%.

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

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 average 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 average 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.

Risk Impact related to 10-year Test Interval

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 is obtained by applying the 3.33 multiplier to the base values as shown below:

$FREQ_{Class3A10}$ = 9.45E-07 * 3.33 = 3.15E-06 / year
$FREQ_{Class3B10}$= 9.45E-08 * 3.33 = 3.15E-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-1-$FREQ_{10}$ = $FREQ_{Class-1}$ - ($FREQ_{Class3A}$ + $FREQ_{Class3B}$)

CLASS-1-$FREQ_{10}$ = 1.76E-05 - (3.15E-06 + 3.15E-07)

CLASS-1-$FREQ_{10}$ = 1.41E-05 / 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	1.41E-05	4.24E+02	5.98E-03
2	Large Isolation Failures (Failure-to-close)	6.45E-06	1.85E+06	11.93
3A	Small Isolation Failures (Type A test)	3.15E-06	4.24E+03	1.34E-02
3B	Large Isolation Failures (Type A test)	3.15E-07	1.48E+04	4.66E-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)	4.99E-06	5.68E+04	0.28
8	Bypass (SGTR)	5.89E-06	4.26E+06	25.09
CDF	All CET End States	**3.50E-05**	N/A	**37.32**

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 = 1.34E-02person-rem/year [Table 4]

$CLASS3B_{10}$ = Class 3B person-rem/year = 4.66E-03person-rem/year [Table 4]

$Total_{10}$ = total person-rem year for 10-year interval = 37.32person-rem/year [Table 4]

$$\% \text{ of Total } DR_{10} = [(CLASS3A_{10} + CLASS3B_{10}) / Total_{10}] \times 100$$
$$\% \text{ of Total } DR_{10} = [(1.34E\text{-}02 + 4.66E\text{-}03) / 37.32] \times 100$$
$$\textbf{\% of Total } \mathbf{DR_{10}} = 0.048\%$$

Therefore, the total 10-year test frequency ILRT interval percent of total dose rate related to Type A testing is 0.048%.

The Δ% change in the 10 year ILRT Dose Rate from the baseline value is 0.048% - 0.015% = 0.033%.

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

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

where:

Σ^n (Class 1,3A,3B)$_{Base}$ = 7.04E-03 + 4.01E-03 + 1.40E-03 person-rem/year [Table 3]

Σ^n (Class 1,3A,3B)$_{Base}$ = 1.25E-02person-rem/year

Σ^n (Class 1,3A,3B)$_{10}$ = 5.98E-03 + 1.34E-02 + 4.66E-03 person-rem/year [Table 4]

Σ^n (Class 1,3A,3B)$_{10}$ = 2.40E-02person-rem/year

DR Change$_{10}$ = [2.40E-02 - 1.25E-02] person-rem/year
DR Change$_{10}$ = 1.15E-02 person-rem/year

Therefore, the ten-year dose rate change from the baseline case is 1.15E-02person-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$_{Class\ 3A15}$ = 9.45E-07 * 5.0 = 4.73E-06 / year
FREQ$_{Class\ 3B15}$ = 9.45E-08 * 5.0 = 4.73E-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-1-FREQ$_{15}$ = FREQ$_{Class-1}$ - (FREQ$_{Class3A}$ + FREQ$_{Class3B}$)

CLASS-1-FREQ$_{15}$ = 1.76E-05 - (4.73E-06 + 4.73E-07)

CLASS-1-FREQ$_{15}$ = 1.24E-05 / year

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	1.24E-05	4.24E+02	5.26E-03
2	Large Isolation Failures (Failure-to-close)	6.45E-06	1.85E+06	11.93
3A	Small Isolation Failures (Type A test)	4.73E-06	4.24E+03	2.01E-02
3B	Large Isolation Failures (Type A test)	4.73E-07	1.48E+04	7.00E-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)	4.99E-06	5.68E+04	0.28
8	Bypass (SGTR)	5.89E-06	4.26E+06	25.09
CDF	All CET End States	**3.50E-05**	N/A	**37.33**

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

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

where:

$CLASS3A_{15}$ = Class 3A person-rem/year = 2.01E-02person-rem/year [Table 5]

$CLASS3B_{15}$ = Class 3B person-rem/year = 7.00E-03person-rem/year [Table 5]

$Total_{15}$ = total person-rem/year for 15-year interval = 37.33person-rem/year [Table 5]

% of Total DR_{15} = [(2.01E-02 + 7.00E-03) / 37.33] x 100
% of Total DR_{15} = 0.073%

Therefore, the total 15 year test frequency ILRT interval percent of total dose rate related to Type A testing is 0.073%.

The Δ% change in the 15 year ILRT DR from the baseline value is 0.073% - 0.015% = 0.058%.

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

Δ% Change$_{10\text{-}15}$ = % of Total DR$_{15}$ - % of Total DR$_{10}$ = 0.073% - 0.048% = 0.025%

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

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

where:

Σ^n (Class 1, 3A, 3B)$_{\text{Base}}$ = 7.04E-03 + 4.01E-03 + 1.40E-03 person-rem/year [Table 3]

Σ^n (Class 1, 3A, 3B)$_{\text{Base}}$ = 1.25E-02person-rem/year

Σ^n (Class 1, 3A, 3B)$_{15}$ = 5.26E-03 + 2.01E-02 + 7.00E-03 person-rem/year [Table 5]

Σ^n (Class 1, 3A, 3B)$_{15}$ = 3.24E-02person-rem/year

DR Change$_{15}$ = [3.24E-02 - 1.25E-02] person-rem/year

DR Change$_{15}$ = 1.99E-02 person-rem/year

Therefore, the fifteen-year dose rate change from the baseline case is 1.99E-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 same leak size and the corresponding leak rate data was used for North Anna as was used for Surry [CALC08] since the containment size and design pressure is approximately the same. 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).

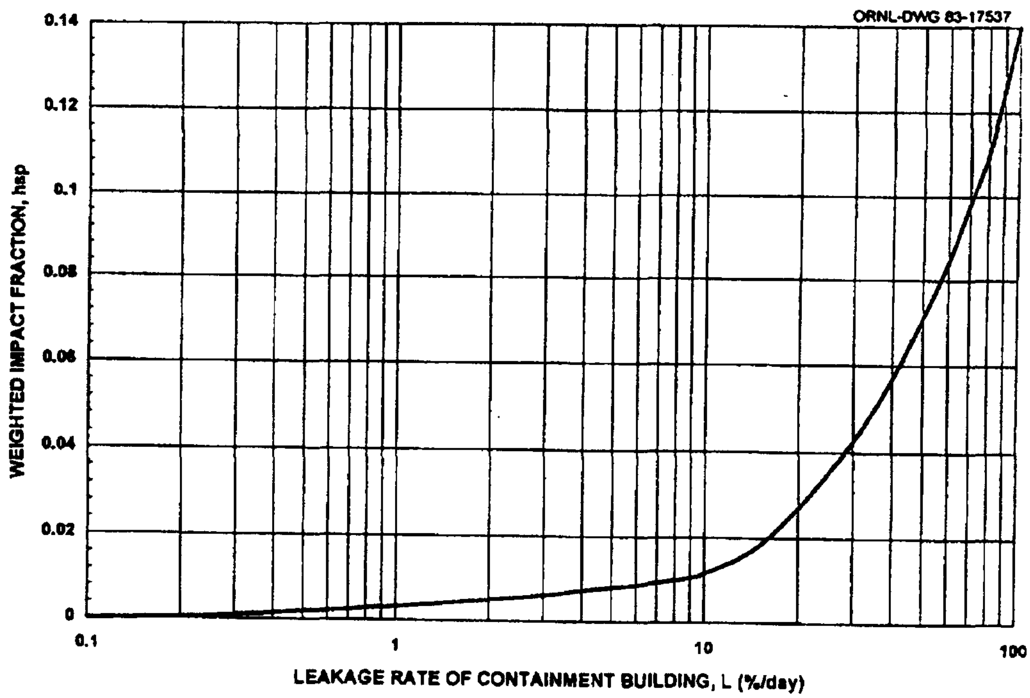
Based upon the information in Table 6 and ORNL, it is judged that small leaks resulting from a severe accident (that are deemed not to dominate public risk) can be defined as those that change risk by less than 5%. This definition would include leaks of less than 35%/day. Based on the Table 6 data, a 35%/day containment leak rate equates to a diameter leak of slightly smaller than 0.7 inches. It is to be noted that for North Anna 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 North Anna 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 North Anna 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 North Anna. This frequency, based on a ten-year test interval, is 3.15E-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 = 3.15E-07 – 9.45E-08 = 2.21E-07

The baseline total LERF for North Anna has been calculated to be 8.20E-07 / yr in [NB01]. This Δ LERF increases the baseline LERF to 8.20E-07 + 2.21E-7 = 1.04E-06 / 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 = 4.73E-07 – 9.45E-08 = 3.79E-07

This Δ LERF increases the baseline LERF to 8.20E-07 + 3.79E-07 = 1.20E-06 / 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 = 4.73E-07 – 3.15E-07 = 1.58E-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 NAPS has been calculated to be 1.20E-06 / 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 (1.58E-07) is in the range of 10^{-7}/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]:

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

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

$$CCFP_{Base} = [1 - ((1.66E\text{-}05 + 9.45E\text{-}07) / 3.50E\text{-}05)] \times 100\%$$
$$CCFP_{Base} = 49.9\%$$

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

$$CCFP_{10} = [1 - ((1.41E\text{-}05 + 3.15E\text{-}06) / 3.50E\text{-}05)] \times 100\%$$
$$CCFP_{10} = 50.7\%$$

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

$$CCFP_{15} = [1 - ((1.24E\text{-}05 + 4.73E\text{-}06) / 3.50E\text{-}05)] \times 100\%$$
$$CCFP_{15} = 51.1\%$$

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

$$\Delta CCFP_{\%} = CCFP_{15} - CCFP_{10} = 0.40\%$$

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

$$\Delta CCFP_{\%} = CCFP_{10} - CCFP_{Base} = 0.80\%$$

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

$$\Delta CCFP_{\%} = CCFP_{15} - CCFP_{Base} = 1.20\%$$

This 15 year change in $CCFP_{\%}$ is slightly greater than 1 percent, and it is considered to be small from a risk perspective.

External Event Sensitivity Analysis

The NAPS 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 NAPS 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 = 9.45E-08 x 2.0 = 1.89E-07 /yr
15 year Class 3B frequency = 4.73E-07 x 2.0 = 9.46E-07 /yr

The external events change in LERF from the Baseline to the 15 year test interval is 7.57E-07 /yr (9.46E-07 – 1.89E-07). This compares to the internal events Baseline to 15 year change in LERF as 3.79E-07 /yr (4.73E-07 – 9.45E-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 as a value between 10^{-7}/yr and 10^{-6}/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 were limited to *5.5* years to reflect the years since September 1996 when 10 CFR *50.55a* started requiring visual inspection. Additional success data were not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date and 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** During 15-year interval, assume failure rate doubles every five years (i.e. a 14.9% increase per year). The average over the 5th through 10 th 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).	**Year** 0 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15	**Flaw Likelihood** 1.79E-03 2.05E-03 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	**Year** 0 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15	**Flaw Likelihood** 4.47E-04 5.13E-04 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 □LERF 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 □LERF 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	Pressure (psia) 20 **64.7 (ILRT)** 100 120	Likelihood of Breach 0.1% **1.1%** 7.02% 20.3%	Pressure (psia) 20 0.01% **64.7 (ILRT)** 100	Likelihood of Breach **0.11%** 0.7%

	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. The same value will be used for NAPS as was used for CCNP, since the containment design is somewhat similar. The design pressure of NAPS is 45 psig versus 50 psig for CCNPP.	150 100%	120 2.0% 150 10%
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-LERF containment over-pressurization failure for NAPS is estimated at 1.24E-05 per year. This is based on the total CDF minus the Class 1,3B and 8 frequencies from Table 1 (1.24E-05 = 3.50E-05 – (1.66E-05 + 9.45E-08 + 5.89E-06)). The total CDF for NAPS is 3.50E-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 * 1.24E-05 = 1.64E-09 per year

Thus it has been independently shown that the increase in LERF due to a liner corrosion failure is 1.64E-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 once-per-ten-years to a once-per-fifteen-years is 1.58E-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	1.76E-05	1.66E-05	4.24E+02	7.04E-03	1.41E-05	5.98E-03		1.24E-05	5.26E-03	
2	6.45E-06	6.45E-06	1.85E+06	11.93	6.45E-06	11.93		6.45E-06	11.93	
3A	N/A	9.45E-07	4.24E+03	4.01E-03	3.15E-06	1.34E-02		4.73E-06	2.01E-02	
3B	N/A	9.45E-08	1.48E+04	1.40E-03	3.15E-07	4.66E-03		4.73E-07	7.00E-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	4.99E-06	4.99E-06	5.68E+04	0.28	4.99E-06	0.28		4.99E-06	0.28	
8	5.89E-06	5.89E-06	4.26E+06	25.09	5.89E-06	25.09		5.89E-06	25.09	
TOTAL	**3.50E-05**	**3.50E-05**	N/A	**37.31**	**3.50E-05**	**37.32**		**3.50E-05**	**37.33**	
	(CDF)	(CDF)			(CDF)			(CDF)		
ΔDR Change from Base							1.15E-02			1.99E-02
ILRT DR % of total dose				0.015			0.048			0.073
Δ% Change in ILRT DR from Base							0.033			0.058
LERF				8.20E-07			1.04E-06			1.20E-06
Δ LERF From base							2.21E-07			3.79E-07
CCFP, %				49.9			50.7			51.1
Δ CCFP, %							0.80			1.2
CDF	**3.50E-05**									

6.0 REFERENCES

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[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

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ATTACHMENT A, NAPS CORE DAMAGE FREQUENCY AND DOSE DATA

Table A-1: NAPS Frequency and Dose Data

Release Category	Frequency* Per year	Person-Rem**	EPRI Class	Description
1	1.76E-05	+4.24E+02	1	No CF
2	0.00E+00	4.24E+02	7	Early CF
3	0.00E+00	++6.69E+05	7	Early CF
4	0.00E+00	++1.45E+06	7	Early CF
5	0.00E+00	6.69E+05	7	Early CF
6	0.00E+00	++1.45E+06	7	Early CF
7	0.00E+00	2.35E+06	7	Early CF
8	0.00E+00	1.45E+06	7	Early CF
9	6.95E-07	++1.99E+04	7	Late CF
10	4.51E-08	++6.69E+05	7	Late CF
11	3.84E-08	1.99E+04	7	Late CF
12	3.85E-08	++6.69E+05	7	Late CF
13	4.24E-10	2.68E+05	7	Late CF
14	2.22E-06	++5.60E+04	7	Late CF
15	1.38E-06	5.60E+04	7	Late CF
16	5.77E-07	++1.99E+04	7	Melthru
17	6.12E-08	++4.24E+02	2	No Cont. Iso
18	4.10E-08	3.86E+05	2	No Cont. Iso
19	0.00E+00	++1.45E+06	2	Alpha CF
20	6.35E-06	+++9.54E+03	2	Debris Cool IV
21	2.04E-08	-------	1	Debris Cool IV
22	1.36E-06	2.41E+06	8	V-Sequence
23	2.40E-07	6.15E+06	8	V-Sequence
24	4.29E-06	4.74E+06	8	SGTR
CDF Freq	3.50E-05			

* Frequency data taken from [CALC02].
**Person-Rem data taken from [CALC01].
+ Used same dose as STC 2 (MAAP run has characteristics that are representative of an EPRI Class 1 containment leakage).
++ Recommended Alternate values were used consistent with the IPE and SAMA analysis
+++ Use IPE STC 20 instead of STC 21 based on review of MAAP runs.
Total Class 1 Frequency = 1.76E-05 yr^{-1}
Total Class 2 Frequency = 6.45E-06 yr^{-1}
Total Class 7 Frequency = 4.99E-06 yr^{-1}
Total Class 8 Frequency = 5.89E-06 yr^{-1}

ATTACHMENT B, JUSTIFICATION OF VOLUME CHANGE

Revision 1

Editorial revision for enhancement and clarity prior to NRC LAR submittal

ATTACHMENT C, REVIEWER COMMENTS/RESOLUTIONS

Comment Number	Section /Page	Review Comment	Response to Review Comment
1	All	Minor Editorial Comments	Corrected
2	All	Minor Editorial Comments	Corrected

Attachment 3

Marked-up Technical Specifications Change

North Anna Power Station Unit 2
Virginia Electric and Power Company
(Dominion)

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

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

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

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

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exception:

NEI 94-01-1995, Section 9.2.3: The first Unit 2 ~~1~~ Type A test performed after the October 9, 1999 ~~April 3, 1993~~ Type A test shall be performed no later than October 9, 2014 ~~April 2, 2008~~.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42.7 psig. The containment design pressure is 45 psig.

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

(continued)

Attachment 4

Proposed Technical Specifications Change

North Anna Power Station Unit 2
Virginia Electric and Power Company
(Dominion)

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

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

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

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

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

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   NEI 94-01-1995, Section 9.2.3: The first Unit 2 Type A test performed after the October 9, 1999 Type A test shall be performed no later than October 9, 2014.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42.7 psig. The containment design pressure is 45 psig.

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

(continued)