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June 14, 2006

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

Serial No.	06-315
NSS&L/DF	R0
Docket No.	50-423
License No.	NPF-49

DOMINION NUCLEAR CONNECTICUT, INC. (DNC)
MILLSTONE POWER STATION UNIT 3
PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE
FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL (LBDCR 06-MP3-010)

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) requests an amendment to Facility Operating License Number NPF-49 in the form of a change to the technical specifications for Millstone Power Station Unit 3 (MPS3). The proposed change will permit a one-time, five-year extension of the ten-year performance-based Type A test interval established in NEI 94-01, "Nuclear Energy Institute Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, dated July 26, 1995. The risk assessment methodology used to support this amendment is based on EPRI's "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," which was developed for the Nuclear Energy Institute in December 2003.

This change has been prepared in accordance with the guidance provided in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk Informed Decisions on Plant Specific Changes to the Licensing Basis." A discussion of the proposed change and the associated supporting risk assessment are included in Attachments 1 and 2 of this letter, respectively. A mark-up of Technical Specification 6.8.4.f, "Containment Leakage Rate Testing Program," is provided in Attachment 3. The retyped technical specification page is provided in Attachment 4.

The proposed amendment does not involve a significant impact on public health and safety and does not involve a significant hazards consideration pursuant to the provisions of 10 CFR 50.92 (see Significant Hazards Consideration in Attachment 1).

The Site Operations Review Committee has reviewed and concurred with the determinations.

To permit effective Cycle 12 planning, DNC is requesting NRC staff review and approval of the proposed change by February 2007. Once approved, the amendment will be implemented within 60 days.

Should you have any questions or require additional information, please contact Mr. Paul R. Willoughby at (804) 273-3572.

Very truly yours,

Eugene S. Grecheck
Vice President – Nuclear Support Services

Attachments (4)

1. Evaluation of Proposed License Amendment
2. Risk Impact Assessment
3. Mark-up of Technical Specifications
4. Retyped Technical Specifications Page

Commitments made in this letter: None

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COMMONWEALTH OF VIRGINIA)
)
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The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck, who is Vice President – Nuclear Support Services, of Dominion Nuclear Connecticut, Inc. 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 14th day of June, 2006.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

(SEAL)

ATTACHMENT 1

PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL (LBDCR 06-MP3-010)

EVALUATION OF PROPOSED LICENSE AMENDMENT

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

EVALUATION OF PROPOSED LICENSE AMENDMENT

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) requests a change to the surveillance requirements referenced in Section 4.6.1 of the Millstone Power Station Unit 3 technical specifications for the containment structure. The proposed change will permit Millstone Power Station Unit 3 (MPS3) a one-time, five-year extension to the requirement of NEI 94-01 (Reference 1) which specifies performance of an integrated leak rate test (ILRT) at a frequency of up to ten years with allowance for a fifteen-month extension.

A mark-up of Technical Specifications page 6.17 incorporating the proposed change to Technical Specification 6.8.4.f, "Containment Leakage Rate Testing Program," is provided in Attachment 3 and the retyped page is provided in Attachment 4. The wording in MPS3 Technical Specification 4.6.1.2 will remain the same.

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.

2.0 PROPOSED CHANGE

The proposed change will modify the first paragraph of Technical Specification 6.8.4.f as follows:

Current

> A program shall be established to implement the leakage rate testing on 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 (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

Proposed

> A program shall be established to implement the leakage rate testing on 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 RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Performance-Based Option of 10 CFR Part 50, Appendix J": The first Type A test performed after the January 6, 1998 Type A test shall be performed no later than January 6, 2013.

3.0 BACKGROUND

The MPS3 current ten-year Type A test interval ends on January 6, 2008. In order to meet the interval requirements of NEI 94-01, this test must be performed during either the spring 2007 refueling outage, or using the fifteen-month extension provision, during the fall 2008 refueling outage.

The proposed amendment to the TS takes a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A test as documented by NEI 94-01. The exception will allow ILRT testing within 15 years from the last ILRT which was performed in January 1998.

The proposed changes discussed within this license amendment request are similar to license amendments issued to Millstone Power Station Unit 2 License No. DPR-65 (Amendment No. 285) on April 6, 2005, Surry Power Station Unit 1 License No. DPR-32 (Amendment No.233) on December 16, 2002, and to North Anna Power Station Unit 1 License No. NPF-4 (Amendment No. 234) on December 31, 2002, and Vermont Yankee Nuclear Power Station License No. DPR-28 (Amendment No. 227) on August 31, 2005.

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.

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 186 (Reference 2) was issued to Millstone Power Station Unit 3 to permit implementation of 10 CFR 50, Appendix J, Option B. Amendment 186 modified Technical Specification Section 4.6.1 and Technical Specification 6.8.4 which require testing in accordance with the Containment Leakage Testing Program and RG 1.163 (Reference 3), respectively. RG 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8–1994 (Reference 4), subject to several regulatory positions in the guide.

The adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but it did alter the frequency of measuring primary containment leakage in Type A, B and C tests. Frequency is based upon an evaluation which looks at the "as

found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The changes to the Type A test frequency did not result in an increase in containment leakage. Similarly, this proposed change to the Type A test frequency will not result in an increase in containment leakage.

3.2 Reason for Proposed Amendment

The frequency interval for testing allowed by 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 (ILRT) 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 ILRTs 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 ILRT testing had minimal impact on public risk."

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

Exceptions to the requirements of RG 1.163, are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states, "The Regulatory Guide or other implementing document used by a licensee or applicant for an operating license, to develop a performance based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide." Since exceptions meeting the stated requirements are permitted, license amendment requests satisfying these requirements do not require an exemption to Option B.

4.0 TECHNICAL ANALYSIS

4.1 Plant Specific Risk Assessment for the Extended ILRT Test Interval

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 leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3. Based on the

October 1993 and January 1998 ILRTs, the current interval for MPS3 is once every ten years.

A risk assessment was performed in accordance with the guidelines set forth in NEI 94-01, the methodology used in EPRI TR-104285 (Reference 6), the NEI Interim Guidance (Reference 7), 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, RG 1.174 (Reference 8). The risk impact assessment is provided in Attachment 2 of this letter.

4.1.1 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.

4.1.2 Conclusions

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

- 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 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF.

  The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once-per-fifteen-years is 3.1×10^{-7}/yr based on internal events. RG 1.174 states that when the calculated increase in LERF is in the range of 10^{-7}/yr to 10^{-6}/yr, applications will be considered if it can be shown that the total LERF is less than 10^{-5}/yr. Since the total LERF for the 15-year metric is 6.3×10^{-7}/yr, then the proposed change is considered acceptable.

- The increase in the total integrated plant risk 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 ILRT dose rate by 2.1%. This change in dose rate is due to the conservative assumption made in the source term release fraction calculation.

- The change in conditional containment failure probability (CCFP) is calculated to demonstrate the impact on 'defense-in-depth.' For the current ten-year ILRT interval, the contribution of sequences involving containment failure for the ten-year interval is 50.2%. For the proposed fifteen-year interval, the contribution of sequences involving containment failure increased to 50.7%. Therefore, the $\Delta CCFP_{10-15}$ is found to be 0.5%. This represents a small change in the MPS3 containment defense-in-depth.

4.2 10 CFR 50 Appendix J, Option B Integrated Leak Test Information

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 Millstone Power Station Unit 3 reactor containment structure has extremely low leakage and represents an 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 inservice 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 two Type A tests are reported in the following table for MPS3:

Date	As Found Leakage Rate(*) WT%/day	Limit(**) WT%/day	Test Pressure (psig)
October 12, 1993	0.1333	0.65	39
January 6, 1998	0.1158	0.3	39

* This is the leakage attributable to the leakage integrity of the containment structure. It is calculated as the sum of the Type A upper confidence limit (UCL) and As-Left minimum pathway leakage for all Type B and Type C pathways in service, isolated and not lined up in their test position prior to performing the test.

** The performance criteria for Type A test allowable leakage is less than 1.0 L_a

4.3 Plant Operational Performance

During power operation, control room instrumentation provides constant indication of containment pressure. With the containment at subatmospheric conditions, if pressure rises, an alarm annunciates advising conditions are approaching the limits allowed by the technical specifications. Additionally, any significant degradation would become evident by the inability to maintain containment vacuum or by excessive operation of the containment vacuum pumps. This monitoring of the containment pressure equates to continuous on-line monitoring of the containment leakage during operation.

4.4 IWE/IWL Inservice Inspection (ISI) Program and Activities to Support ILRT

The current regulatory requirement mandated by 10 CFR 50.55a requires licensees to implement a containment inspection program in accordance with the rules and requirements of the 1992 Edition through the 1992 Addenda of ASME Section XI, Subsections IWE and IWL, as amended in the regulation. DNC implemented the Containment ISI Program in accordance with these rules at each of its two operating nuclear units. The regulatory requirement allows five years for the implementation of the first period inspections. In consideration of these rules, the Initial Period (First Period) for the performance of Containment ISI began on September 9, 1996 and ended on September 9, 2001. The subsequent periods (IWE) comply with the normal period requirements of four years for the second period and three years for the third period of inspection program B of ASME Section XI. The subsequent IWL intervals are repeated every five years. The proposed frequency extension of ILRT requirements would have no effect upon these requirements. The regulation requires the general visual

examination, IWE Category E-A, be conducted each inspection period during the interval in addition to the Code requirement that is to be completed just prior to the Type A test. This general visual examination requirement is similar to the visual requirement of Appendix J. The general visual examination requirement conducted each period will be maintained during the extended ILRT period beyond the normal code required ten-year interval. No Code requirement (IWE, Category E-A) will be affected by the ILRT period extension.

The following relief requests were reviewed to assess the effect, if any, resulting from the proposed ILRT period extension:

- Relief Requests RR-E1 and RR-L1 requested relief from Section XI of the ASME Code, 1992 Edition, 1992 Addenda, for all IWE and IWL zones, respectively. The relief permits the use of the rules provided in the ASME Code Section XI, 1998 Edition, Subsections IWE and IWL for Class MC and Class CC examinations required to be performed under the expedited containment examination rules of 10 CFR 50.55a(g)(6)(ii)(B). The NRC letter dated April 21, 2000, granted this relief to Millstone Unit 3 (Reference 9). The proposed ILRT period extension only affects the length of time between Type A testing. The type or method of examinations is not changed and, therefore, the relief request remains valid and unaffected by the proposed change.

- Relief Request RR-E2 requested relief from Section XI of the ASME Code, 1998 Edition, IWE-2500(b)(1) which requires detailed visual examination of both sides of an accessible surface and IWE(b)(2) which requires ultrasonic thickness measurements. DNC's relief request proposed the use of detailed visual examination on the accessible surface areas supplemented by volumetric examination as specified as part of the engineering evaluation of each E-C category surface. The NRC letter dated November 14, 2000, granted this relief to Millstone Power Station Unit 3 (Reference 10). The proposed ILRT frequency extension affects Type A testing only. The examination methods authorized by the NRC remain unchanged. As a result, the relief request remains valid and is unaffected by the proposed change.

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

- During the March 2001 and April 2004 refueling outages, DNC performed IWE General Visual examinations of the Containment Metal Liner (IWE - MC component). All accessible surface areas were examined including the area of the liner behind the 1/2-inch by 1/2-inch moisture barrier seal. Some localized rust and surface anomalies were detected, most associated near or at attachments to the upper uncoated dome liner area or behind the moisture

barrier seal. Repairs scheduled for a number of Examination Category E-C (accelerated degradation) classified items were made during the past two refueling outages. They included repairs to the moisture barrier seal and recoating of some of the liner surface area at selected localized rust areas. A number of new E-C items, most located in the upper dome area, were identified in the April 2004 examination. They were conservatively classified as E-C to the next inspection period and are expected to be reclassified the next inspection.

- Inspections of the containment liner are performed during the interval between ILRTs. The extension of the ILRT period will not affect the inspections.

- The performance-based ILRT program guidance (NEI 94-01 and Regulatory Guide 1.163) requires a minimum of three inspections of the accessible portions of the inside and outside of the containment structure to assess the condition of the containment structure during the ten year interval. Engineering personnel performed these inspections to an owner-defined program up through the March 2001 refueling outage. Certified Level II VT-1 and VT-3 visual examiners performed these inspections under the supervision of the Responsible Engineer during the April 2004 refueling outage. Any identified discrepancies noted in the liner, penetrations or concrete are documented and dispositioned in accordance with the appropriate Code/design requirements. These inspections are conducted using a mixture of direct and remote examination techniques.

- The accessible portions of the containment liner are inspected during each of the three periods in the ten-year inspection interval as required by ASME Code, Section IWE. These inspections are performed by qualified personnel, and any identified discrepancies are documented and dispositioned in accordance with ASME Section XI requirements. These inspections are conducted using a mixture of direct and remote examination techniques.

- Coating inspections are performed each outage on accessible portions of the containment liner by engineering personnel. Any identified discrepancies in the coating or liner are documented and dispositioned in accordance with the appropriate design standards.

The above visual inspections of the containment have proven to be effective in identifying degradation of either the interior liner or the exterior concrete surface.

4.5 Safety Related Porous Concrete Groundwater Sump (Underdrain System Sump)

There is a porous concrete groundwater sump and non-safety related sump pump located in the engineered safety features building that collects (via an underdrain and porous concrete media) any significant amount of groundwater seepage which has circumvented the waterproof membrane. The sump protects the containment steel liner from hydrostatic loading. The electric sump pump is powered from a non-safety related

electrical circuit which derives power from a safety-related electrical bus, providing greater assurance of a reliable energy source. (FSAR Section 3.8.1.6.4 contains information on the waterproof membrane, and FSAR Section 9.3.3.2.4.1 contains additional details on the sump pump.)

4.6 Containment Liner Corrosion Sensitivity Analysis

An undetected through-wall hole in both the concrete and the liner, at approximately the same location would have to be postulated to be a LERF contributor. Furthermore, both leak paths would have to exist long enough for the pathways to grow sufficiently such that the release would be large enough to be considered a LERF contributor. As a result of the liner and concrete inspections, the likelihood of an undetected through-wall path from the containment atmosphere to the environment for even a very small leak is considered to be remote. The likelihood of occurrence of an undetected through-wall path becomes even smaller as the assumed leak size increases. A sensitivity analysis has been performed to estimate the impact of failure from a defect initiated between the containment wall and the liner. This sensitivity analysis used historical data to establish flaw likelihood. Given the assumed liner flaw, the containment fragility analysis is used to estimate the probability of breaching the containment at the design pressure. Finally, the likelihood of visual detection failure is assessed and included in the analysis. The product of these terms is the likelihood of non-detected containment leakage, which was calculated for both the containment cylinder and the basemat in the sensitivity analysis. The product of this likelihood and the non-large early release frequency is the increase in LERF due to non-detected containment leakage. The key calculations and assumptions in the sensitivity analysis are located in Attachment 2.

4.7 Fuel Transfer Tube Bellows

The MPS3 fuel transfer tube consists of a sleeve welded to the containment liner and attached to the transfer tube by means of a bellows connection. The area between the tube and the sleeve is provided with a test connection for testing the bellows seal connection. The fuel transfer tube blind flange is double-gasketed and can be pressurized for Type B leakage rate testing each refueling outage.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed revision to the Millstone Power Station Unit 3 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, would be extended on a one-time basis to fifteen years from the last Type A test for Millstone Power Station Unit 3.

Dominion Nuclear Connecticut, Inc. (DNC) has evaluated whether or not a Significant Hazards Consideration is involved with the proposed changes by addressing the three standards set forth in 10 CFR 50.92(c) as discussed below.

Criterion 1:

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

Response: No

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 it an activity or modification that by itself could lead to equipment failure or accident initiation.

The proposed one-time, five-year 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 even reducing the Type A (ILRT) testing frequency to once per twenty years leads to an imperceptible increase in risk.

DNC provides a high degree of assurance through indirect testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last two Type A tests identified containment leakage within acceptance criteria, indicating a very leak-tight containment. Inspections required by the ASME Code are also performed in order to identify indications of containment degradation that could affect leak-tightness. Separately, Type B and C testing required by Technical Specifications, identifies any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that a one-time, five-year extension to the Millstone Power Station Unit 3 Type A test interval will not represent a significant increase in the consequences of an accident.

Criterion 2:

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

Response: No

The proposed revision to Technical Specifications adds a one-time extension to the current interval for Type A testing for Millstone Power Station Unit 3. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing does not create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure.

Criterion 3:

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

Response: No

The proposed revision to Millstone Power Station Unit 3 Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test for Millstone Power Station Unit 3. 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 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once-per-fifteen-years is 3.1×10^{-7}/yr, based on internal events. RG 1.174 states that when the calculated increase in LERF is in the range of 10^{-7}/yr to 10^{-6}/yr, applications will be considered if it can be shown that the total LEFT is less than 10^{-5}/yr. Since the total LERF for the 15-year metric is 6.3×10^{-7}/yr, then the change is considered acceptable. Increasing the ILRT 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 interval in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG-1493 generically concludes that the design containment leakage rate contributes about 0.1 percent of the overall risk. 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.

In summary, DNC concludes that the proposed amendment does not represent a Significant Hazards Consideration under the standards set forth in 10 CFR 50.92(c).

6.0 ENVIRONMENTAL CONSIDERATION

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

7.0 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 Millstone Unit 3 Issuing Technical Specification Amendment 186, dated November 2, 2000, 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. 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. 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.

9. USNRC letter from James W. Clifford to S.E. Scace, "Safety Evaluation for Alternative Associated With the Use of Subsection IWE and IWL of the ASME Code for Containment Inspection, Millstone Nuclear Power Station, Unit Nos. 2 and 3 (TAC Nos. MA5332 and MA5338)," dated April 21, 2000.

10. USNRC letter from James W. Clifford to S.E. Scace, "Safety Evaluation of Relief Request RR-E2 for the Containment Inspection Program, Millstone Nuclear Power Station, Unit Nos. 2 and 3 (TAC Nos. MB0164 and MB0165)," dated November 14, 2000.

ATTACHMENT 2

PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL (LBDCR 06-MP3-010)

RISK IMPACT ASSESSMENT

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

RISK IMPACT ASSESSMENT

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1. Purpose

Provide a risk impact assessment on extending the plant's integrated leak rate test (ILRT) interval from ten to fifteen years. The risk assessment is performed in accordance with the guidelines set forth in NEI 94-01 [1], the NEI interim guidance [21], the methodology used in EPRI reports [2, 22] and the NRC regulatory guidance RG 1.174 [3]. The RG provides use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis.

2. Summary

In 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 Millstone Unit 3 Nuclear Power Station (MPS3) selected the requirements under Option B as its testing program [4].

The surveillance testing requirements as proposed in NEI 94-01 [1] for Type A testing is at least once per 10 years based on an acceptable performance history (defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than $1L_a$).

The Millstone Unit 3 current ten-year Type A test interval ends on January 6, 2008. The proposed amendment to the TS takes a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A test as documented in NEI 94-01. The exception will allow ILRT testing within 15 years from the last ILRT which was performed in January 1998.

This calculation will provide a risk impact assessment on extending the plant's integrated leak rate test (ILRT) interval from ten to fifteen years. The risk assessment will be performed in accordance with the guidelines set forth by NEI [1, 21], the methodology used by EPRI [2, 22] 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, RG 1.174 [3].

In addition, the results and findings from the Millstone Individual Plant Examination (IPE) [5], the level 2 data from the license renewal SAMA [23] and the revised model [10,17] are used for this risk assessment calculation.

3. References

1) RF-Report, NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 26, 1995, Revision 0.
2) RF-Report, EPRI TR-104285, "Risk Assessment of Revised Containment Leak Rate Testing Intervals," August 1994.

3) RF-Report, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," July 1998.
4) RF-Procedure, Surveillance Procedure, SP31103, "Containment Integrated Leak Rate Test, Type A," Revision 3, 1998.
5) RF-Report, "Millstone Unit No. 3 Individual Plant Examination For Severe Accident Vulnerabilities," August 1990.
6) RF-Calc., PRA02NQA-01895S3 Revision 1, "MACCS2 model for Millstone Unit 3 Level 3 Application," August 2004.
7) RF-Report, NUREG-1493, "Performance-Based Containment Leak-Test Program," July 1995.
8) RF-Report, United States Nuclear Regulatory Commission, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August 1989.
9) RF-Report, Z. T. Mendoza, et al., "Generic Framework for Individual Plant Examination (IPE) Backend (Level 2) Analysis, Volume 1 - Main Report and Volume 3 - BWR Implementation Guidelines," prepared by SAIC International, Inc., Electrical Power Research Institute, NSAC-159, EPRI PR3114-29, 1991.
10) RF-Calc., PRA02YQA-01822S3, Revision 0, "Millstone 3 PRA Model," October 2002.
11) RF-Report, ERC:25212-ER-04-0015, "MP3 Containment Risk Significant Valve Review", 3-17-2004.
12) RF-Calc., Indian Point 3, IP3-CALC-VC-03357 Revision 0, "Risk Impact Assessment of Extending Containment Type A Test Interval," 1-4-01.
13) RF-Report, Patrick D. T. O'Connor, "Practical Reliability Engineering," John Wiley & Sons, 2nd Edition, 1985.
14) RF-Report, Burns, T.J., "Impact of Containment Building Leakage on LWR Accident Risk," Oak Ridge National Laboratory, NUREG/CR-3539, April 1984.
15) RF-Report, United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.
16) RF-Calc., Florida Power Calculation, F-01-0001, Revision 2, "Evaluation of Risk Significance of ILRT Extension," 6-19-01.
17) RF-Calc., PRA02NQA-01941S3, Revision 2, "MP3 SAMA Impact on Containment Release Frequencies," August 2004.
18) RF-Memo., NSE-06-014, "MP3 ILRT 5-Year Exemption Support," March 23, 2006.
19) RF-Report., 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.
20) RF-Calc., SM-1237, Revision 0, "Surry and North Anna Containment Isolation Modeling," April 20, 2000.
21) RF-Report, "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," Developed for NEI by EPRI, November 2001.
22) RF-Report, EPRI TR-109325, "Risk Impact Assessment of Extended Leak Rate Testing Intervals," December 2003.
23) RF-Report, "Application for Renewed Operating Licenses, Appendix E- Environmental Report, Millstone Power Station, Units 2 and 3," January 2004.

4. Assumptions

The MPS3 leakage rate (L_a) acceptance criteria is defined as:
L_a = 0.3 percent by weight of containment air per 24 hours at calculated peak pressure (P_a).

1. Containment leak rates greater than $1L_a$, but less than 35 L_a, indicate an impaired containment. Break openings of greater than 0.1-inch and less than 0.78-inch in diameter are considered as small leak rate releases. Break openings of greater than 0.78-inch diameter are considered as large leak rate releases.
2. Containment leak rates greater than 35 L_a indicate a containment breach. This leak rate is considered "large".
3. Containment leak rates less than $1L_a$ indicate an intact containment. This leak rate is considered as "negligible".
4. The maximum containment leakage for Class 1 sequences is $1L_a$.
5. The maximum containment leakage for Class 2 sequences is $35L_a$.
6. The maximum containment leakage for Class 3a sequences is $10L_a$.
7. The maximum containment leakage for Class 3b sequences is $35L_a$.
8. The maximum containment leakage for Class 6 sequences is $35L_a$.
9. Because Class 8 sequences are containment bypass sequences, potential releases are directly to the environment. Therefore, the containment structure will not impact the release magnitude.
10. Containment leakage due to Classes 4 and 5 are considered negligible based on the previously approved methodology [12].
11. The containment releases are not impacted with time.
12. 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 Type A.

5. Method of Calculation

A simplified bounding analysis approach for evaluating the change in risk associated with increasing the interval from 10-years-to-15-years for Type A test was used. Type A test measures the containment air mass and calculates the leakage from the change in mass over time. This approach is similar to that presented in EPRI TR-104285 [2], NUREG-1493 [7] and the NEI interim guidance report [21]. Namely, the analysis performed examined the MPS3 IPE [5] 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. For example, liner breach or steam generator manway leakage (EPRI TR-104285 Class 3 sequences). Type B test measures component leakage across pressure retaining boundaries (e.g. gaskets, expansion bellows and air locks). Type C test measures component leakage rates across containment isolation valves.
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. For example, a valve failing to close following a valve stroke test (EPRI TR-104285 Class 6 sequences).
- 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.

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

Step 1 - Quantify the baseline risk in terms of 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 [2]. See Table A-1 of Attachment A.

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

Step 3 - Evaluate risk impact of extending 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 [3].

Step 5 - Evaluate the risk impact in terms of ΔLERF.

Step 6 - Determine impact on conditional containment failure probability.

6. Body of Calculation

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

This step involves the review of the MPS3 IPE [5] 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 [8] and NSAC-159, Volume 2 [9] and on plant features that influence the phenomena.

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing or containment failure induced by severe accident phenomena. 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 valves reported in Reference 11 was made. The five issues associated with containment isolation in NUREG-1335 [8] 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 valves in Reference 11 screened out lines less than 2 inches in diameter. An Expert Panel subcommittee representing Maintenance Rule, Engineering and Operations, evaluated the containment isolation valves at MPS3. The Expert Panel determined that a containment penetration size of 2 inches or less was considered to be non-risk significant.

This ILRT evaluation considers lines between 0.1 inches and 2.0 inches as potential candidates for containment leakage. This is used for the EPRI containment failure classifications. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (i.e., containment liner) exists. The containment leakage for these sequences can be either small ($1L_a$ to $35\ L_a$) or large ($>35\ L_a$).

External event contributors for all accident classes have been considered in this risk assessment. It has been concluded that external events have little impact on the LERF due to the extension of the containment leak rate testing. See further discussion in Section 6 of this calculation.

The Level 3 release categories were mapped into 8 release classes (See Table A-1 in Attachment A) as defined in the EPRI Report [2]. 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 PWR's (e.g. MPS3 measured at P_a, calculated peak containment pressure related to the design basis accident). Changes to leak rate testing frequencies do not affect this classification.

Class 2 Containment isolation failures (as reported in the IPEs) include those accidents in which the pre-existing leakage is due to failure to isolate the containment. These include those that are dependent on the core damage accident in progress (e.g. initiated by common cause failure or support system failure of power) and random failures to close a containment path. Changes in Appendix J testing requirements do not impact these accidents.

Class 3 Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i. e., provide a leak-tight containment) is not dependent on the sequence in progress. This accident class is applicable to sequences involving ILRTs (Type A tests) and potential failures not detectable by LLRTs.

Class 4 Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B- tested components that have isolated but exhibit excessive leakage.

Class 5 Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.

Class 6 Containment isolation failures include those leak paths not identified by the LLRTs. The type of penetration failures considered under this class includes those covered in the plant test and maintenance requirement or verified by inservice 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 3–6 frequencies are calculated first since these values are needed to determine the Class 1 frequency.

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

To calculate the probability that a liner leak will be large (Event CLASS-3B), use was made of the data presented in NUREG-1493 [7]. The data found in NUREG-1493 states that 144 ILRTs were conducted. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Since 21 L_a does not constitute a large release (refer to the write-up in Step 4), no large releases have occurred based on the 144 ILRTs reported in NUREG-1493 [7].

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 Reference 21. The method utilized in the aforementioned utility submittals (discussed in Reference 21) involved using a 95% confidence of a $\chi 2$ distribution of the noted ILRT failures (4 of 144 reported in NUREG-1493). Data collected recently by NEI from 91 nuclear power plants indicates that 38 plants have conducted ILRTs since 1/1/95, with only one failure (due to construction debris from a penetration modification). This would indicate that the statistical information should be based on 5/182. Rather than using the $\chi 2$ distribution used previously, it has been considered more appropriate to utilize the mean (5/182=0.027) for the class 3a distribution.

The Jeffreys non-informative prior distribution (Reference 21) was used for the class 3b distribution:

$$Failure \Pr obability = \frac{NumberofFailures(0) + 1/2}{NumberofTests(182) + 1}$$

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

Compared with the previous revision of this calculation, the impact of the second improvement on the overall results is small, with larger impact for class 3b than for class 3a.

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

Where:
$PROB_{class\text{-}3a}$= probability of small pre-existing containment liner leakage = 0.027

$PROB_{class\text{-}3b}$= probability of large pre-existing containment liner leakage = 0.0027
CDF = 2.88×10^{-5} /year from Reference 17

CLASS-3A-Base-Frequency = 0.027 * 2.88 x 10^{-5} /year = 7.78 x 10^{-7} /year

CLASS-3B-Base-Frequency = 0.0027 * 2.88 x 10^{-5} /year = 7.78 x 10^{-8} /year

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

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.

To be consistent with the NEI interim guidance (Ref. 21) a frequency of zero is used for class 6 accidents.

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 1.46 x 10^{-5} /year (Attachment A, Table A-1). For this analysis the associated maximum containment leakage for this group is $1L_a$. The MPS3 IPE did not model Class 3 type failures, therefore this needs to be accounted for in the Class 1 accident class. Using Reference 21 methodology, 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 = $FREQ_{Class\text{-}1}$ - ($FREQ_{Class3a}$ + $FREQ_{Class3b}$)

CLASS-1-FREQ = 1.46E-05 - (7.78 x 10^{-7} + 7.78 x 10^{-8})

CLASS-1-Base-Frequency = 1.37E-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 as Class 2.

CLASS-2-FREQUENCY = 0.00 /year [Table A-1]

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure 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 as Class 7.

CLASS-7-FREQUENCY = 1.30 x 10^{-5} / year

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

CLASS-8-FREQUENCY = 1.22 x 10^{-6} / 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	1.37E-05
2	Large Containment Isolation Failures (Failure-to-Close)	0.00E+00
3a	Small Isolation Failures (Type A test)	7.78E-07
3b	Large Isolation Failures (Type A test)	7.78E-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)	1.30E-05
8	Containment Bypassed (SGTR & V-Sequence)	1.22E-06
CDF	Core Damage All CET End states	**2.88E-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 MPS3 plant. The no containment failure Class 1 dose was used for Class 1 accident release as shown in Table A-1 in Attachment A. The Source Term Category M-4 containment isolation failure accident sequence has characteristics that are representative of an EPRI Class 2 containment leakage.

Using the total population dose for Class 1 accidents as the starting reference point, the Class 3 through 6 accidents are calculated below. The population dose is converted to the corresponding Class value using the appropriate dose multiplier as was used in Reference 12 to predict the person-rem dose for accident classes 1 to 6 as follows. The dose for the Class 7 accidents was obtained by frequency weighting all the Class 7 dose values and dividing the sum of the products by the sum of the frequencies from Table A-1. The baseline dose results are shown below.

Class 1 = 1.65×10^4 person-rem
Class 2 = 0.00E+00 person-rem
Class 3a = $1.65 \times 10^4 * 10 = 1.65 \times 10^5$ person-rem
Class 3b = $1.65 \times 10^4 * 35 = 5.78 \times 10^5$ person-rem
Class 4 = Not analyzed
Class 5 = Not analyzed
Class 6 = Not analyzed
Class 7 = $\sum^n (\text{Freq} \times \text{Dose}) / \sum^n \text{Freq} = 5.83 \times 10^5$ person-rem
Class 8 = $\sum^n (\text{Freq} \times \text{Dose}) / \sum^n \text{Freq} = 4.10 \times 10^6$ person-rem

Class 8 sequences include containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are expected to be

released directly to the environment. The Class 8 doses are frequency weighted as were done for class 7. The frequency weighted Class 8 dose from Table A-1 represent the sum of the dose for the Event-V and SGTR sequences.

The above values are summarized in Table 2 below.

Table 2
Baseline Person-Rem Measures - Given Accident Class

Class	Description	Person-Rem (50-Miles)
1	No Containment Failure	1.65×10^4
2	Large Containment Isolation Failures (Failure-to-Close)	0.00E+00
3a	Small Isolation Failures (Type A test)	1.65×10^5
3b	Large Isolation Failures (Type A test)	5.78×10^5
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	Other Isolation Failures (e.g., Dependent Failures)	Not Analyzed
7	Failure Induced by Phenomena (Early and Late Failures)	5.83×10^5
8	Containment Bypassed (SGTR & V-Sequence)	4.10×10^6

The above dose results, when combined with the frequency results presented in Table 1, yields the MPS2 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	1.37E-05	1.65×10^4	2.27E-01
2	Large Isolation Failures (Failure-to-Close)	0.00E+00	0.00E+00	0.00E+00
3a	Small Isolation Failures (Type A test)	7.78E-07	1.65×10^5	1.28E-01
3b	Large Isolation Failures (Type A test)	7.78E-08	5.78×10^5	4.49E-02
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)	1.30E-05	5.83×10^5	7.57E+00
8	Containment Bypassed (SGTR & V-Sequence)	1.22E-06	4.10×10^6	5.00E+00
Total	All CET End States	**2.88E-05**	N/A	**12.98**

Based on the above values, using the same methodology as Reference 16, the baseline percent of total dose rate (DR) due to Type A testing is as follows:

% of Total DR_{BASE} =[($CLASS3a_{BASE}$ + $CLASS3b_{BASE}$) / $Total_{BASE}$] X 100

Where:

$CLASS3a_{BASE}$ = class 3a person-rem/year = 1.28×10^{-1} person-rem/year [Table 3]

$CLASS3b_{BASE}$ = class 3b person-rem/year = 4.49×10^{-2} person-rem/year [Table 3]

$Total_{BASE}$ = total person-rem/year for baseline interval = 12.98 person-rem/year [Table 3]

% of Total DR_{BASE} =[(1.28×10^{-1} + 4.49×10^{-2}) / 12.98] x 100%
% of Total DR_{BASE} = 1.3%

Therefore, the baseline percent of total dose rate is 1.3%.

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

The revised methodology in Reference 21 suggests using the following method. It is now believed that the multiplier should be a factor representing the change in probability of leakage. As stated in References 2 and 7, relaxing the test interval from three in ten years to one in ten years increases the average time that a leak detectable only by an ILRT would go undetected from 18 (3 yrs/2) to 60 (10 yrs/2) months. This is a factor of 60/18=3.333. 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 a factor of 90/18 = 5 should be applied.

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

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

The increased leakage for the 10-year Class 3a and 3b frequencies are obtained by applying the 3.33 multiplier to the base values as shown below.

Class 3a = 7.78×10^{-7} *3.33 = 2.59E-06
Class 3b = 7.78×10^{-8} *3.33 = 2.59E-07

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 = $FREQ_{Class\text{-}1}$ - ($FREQ_{Class3a}$ + $FREQ_{Class3b}$)

CLASS-1-FREQ = 1.46×10^{-5} - ($2.59 \times 10^{-6} + 2.59 \times 10^{-7}$)

CLASS-1-FREQ = 1.18E-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	1.18E-05	1.65×10^4	1.94E-01
2	Large Isolation Failures (Failure-to Close)	0.00E+00	0.00E+00	0.00E+00
3a	Small Isolation Failures (Type A test)	2.59 E-06	1.65×10^5	4.27E-01
3b	Large Isolation Failures (Type A test)	2.59 E-07	5.78×10^5	1.50E-01
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)	1.30E-05	5.83×10^5	7.57E+00
8	Bypass (SGTR)	1.22E-06	4.10×10^6	5.00E+00
CDF	All CET End States	**2.88E-05**	N/A	**13.34E+00**

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

% of Total DR_{10} = [($CLASS3a_{10}$ + $CLASS3b_{10}$) / $Total_{10}$] x 100

Where:
$CLASS3a_{10}$ = class 3a person-rem/year = 4.27E-01 person-rem/year [Table 4]

$CLASS3b_{10}$ = class 3b person-rem/year = 1.50E-01 person-rem/year [Table 4]

$Total_{10}$ = total person-rem/year for 10-year interval = 13.34 person-rem/year [Table 4]

% of Total DR_{10} = [(4.27E-01+ 1.50E-01) / 13.34] x 100

% of Total DR_{10} = 4.3%

Therefore, the total 10-year test frequency ILRT interval percent of total dose rate due to Type A testing is 4.3%.

The Δ% change in the 10-year ILRT DR from the baseline value is 4.3% - 1.3% = 3.0%.

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

DR $Change_{10}$ = [Σ^n (Class 1,3a,3b) $_{10}$ - Σ^n (Class 1,3a,3b) $_{Base}$]

Where:
Σ^n (Class 1,3a,3b) $_{Base}$ = 0.40 person-rem/year [Table 3]

Σ^n (Class 1,3a,3b) $_{10}$ = 0.77 person-rem/year [Table 4]

DR $Change_{10}$ = [0.77 - 0.40] person-rem/year
DR $Change_{10}$ = 0.37 person-rem/year

Therefore, the ten-year dose rate change from the baseline case is 0.37 person-rem/year.

Risk Impact due 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 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.

Class 3a = $7.78 \times 10^{-7} * 5$ = 3.89E-06
Class 3b = $7.78 \times 10^{-8} * 5$ = 3.89E-07

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:

$\text{CLASS-1-FREQ} = \text{FREQ}_{\text{Class-1}} - (\text{FREQ}_{\text{Class3a}} + \text{FREQ}_{\text{Class3b}})$

$\text{CLASS-1-FREQ} = 1.46 \times 10^{-5} - (3.89 \times 10^{-6} + 3.89 \times 10^{-7})$

CLASS-1-FREQ = 1.03×10^{-5} /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	1.03E-05	1.65E+04	1.70E-01
2	Large Isolation Failures (Failure-to-Close)	0.00E+00	0.00E+00	0.00E+00
3a	Small Isolation Failures (Type A test)	3.89E-06	1.65E+05	6.42E-01
3b	Large Isolation Failures (Type A test)	3.89E-07	5.78E+05	2.25E-01
4	Small Isolation Failure-to-Seal (Type B test)	N/A	0.00E+00	N/A
5	Small Isolation Failure-to-Seal (Type C test)	N/A	0.00E+00	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	N/A	0.00E+00	N/A
7	Failure Induced by Phenomena (Early and Late Failures)	1.30E-05	5.83E+05	7.57E+00
8	Bypass (SGTR)	1.22E-06	4.10E+06	5.00E+00
CDF	All CET End States	**2.88E-05**	N/A	**13.61E+00**

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

$\% \text{ of Total } DR_{15} = [(\text{CLASS3a}_{15} + \text{CLASS3b}_{15}) / \text{Total}_{15}] \times 100$

Where:
CLASS3a_{15} = class 3a person-rem/year = 6.42E-01 person-rem/year [Table 5]

CLASS3b_{15} = class 3b person-rem/year = 2.25E-01 person-rem/year [Table 5]

Total_{15} = total person-rem/year for 10-year interval = 13.61 person-rem/year [Table 5]

$\% \text{ of Total } DR_{15} = [(6.42\text{E-}01 + 2.25\text{E-}01) / 13.61] \times 100$

$\% \text{ of Total } DR_{15} = 6.4\%$

Therefore, the total 15-year test frequency ILRT interval percent of total dose rate due to Type A testing is 6.4%.

The Δ% change in the 15-year ILRT DR from the baseline value is 6.4% - 1.3% = 5.1%.

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

$\Delta\%\ Change_{10\text{-}15} = \%\ of\ Total\ DR_{15} - \%\ of\ Total\ DR_{10} = 6.4\% - 4.3\% = 2.1\%$

<u>The fifteen-year dose rate change (due to a ILRT) over the baseline case is as follows:</u>

$DR\ Change_{15} = [\Sigma^n\ (Class\ 1{,}3a{,}3b)_{15} - \Sigma^n\ (Class\ 1{,}3a{,}3b)_{Base}]$

Where:
Σ^n (Class 1,3a,3b) $_{Base}$ = 0.40 person-rem/year [Table 3]

Σ^n (Class 1,3a,3b) $_{15}$ = 1.04 person-rem/year [Table 5]

$DR\ Change_{15}$ = [1.04 - 0.40] person-rem/year

$DR\ Change_{15}$ = 0.64 person-rem/year

Therefore, the fifteen-year dose rate change from the baseline case is 0.64 person-rem/year.

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

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

1) The containment leak rate versus breach size, and

2) The impact on risk versus leak rate.

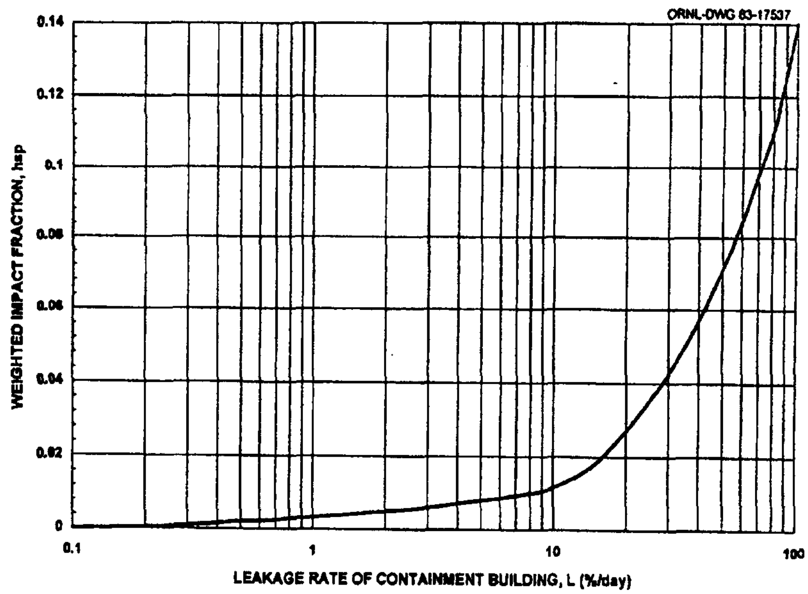
The containment leak size for the corresponding leak rate was calculated using the same methodology as in Reference 20. The effect of containment leak size on the containment leak rate is shown in Table 6. The leak rates were calculated using a containment design pressure of 45psig with the corresponding saturation temperature and a containment free volume of 2.26E6 ft^3. It is to be noted that the MPS3 Technical Specification states that the maximum allowable leakage rate (L_a, at P_a =38.57psig) shall be 0.3% by weight of the containment air per day. In addition, Oak Ridge National Laboratory (ORNL) [14] completed a study evaluating the impact of leak rates on public risk using information from WASH-1400 [15] 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 approximately 0.8 inches. Therefore, this study defines small leakage as containment leakage resulting from an opening of 0.477 in^2 or less, large leakage as greater than 35%/day and negligible leakage as 0.3% /day.

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)**	**Leak Rate (%/day)**
0.072	0.004	0.3
0.132	0.014	1.0
0.417	0.136	10.0
0.780	0.477	35.0
1.317	1.362	100.0
4.166	13.62	1000.0
13.17	136.2	10,000.0

Figure I
Fractional Impact on Risk Associated with Containment Leak Rates [14]



The risk impact associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from containment could in fact result in a 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 $2L_a$). A larger leak rate would imply an impaired containment, such as classes 2, 3, 6 and 7. Since the ILRT does not impact CDF, the relevant metric is LERF.

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 MPS3 IPE [5], 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 MPS3. This frequency, based on a ten-year test interval, is 2.59 x10^{-7}/yr.

Reg. Guide 1.174 [3] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 [3] 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 reasonably shown that the total LERF is less than 10^{-6} 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 Δ LERF

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

Δ LERF = Class $3b_{10}$ – Class $3b_{Base}$

Δ LERF = 2.59E-07 – 7.78E-08 = 1.8E-7

The baseline total LERF for MPS3 has been calculated to be 3.17 $x10^{-7}$/yr in Reference 10. This Δ LERF increases the baseline LERF to 3.17 $x10^{-7}$ + 1.81 $x10^{-7}$ = 5.0 $x10^{-7}$/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 = 3.89E-07 – 7.78E-08 = 3.1E-7

This Δ LERF increases the baseline LERF to 3.17 $x10^{-7}$ + 3.11 $x10^{-7}$ = 6.3 $x10^{-7}$/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 = 3.89E-07 – 2.59E-07 = 1.3E-7

The guidance in Reg. Guide 1.174 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 LERF for MPS3 has been calculated to be 3.17 $x10^{-7}$/yr in Reference 10.

Since guidance in Reg. Guide 1.174 defines small changes in LERF 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 Reg. Guide 1.174 (Ref. 3) 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 (Ref. 21), 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 Reference 21.

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

For the Base interval: The values are obtained from Table 3.

$CCFP_{Base} = [1 - ((1.37E\text{-}05 + 7.78E\text{-}07)/ 2.88E\text{-}05)] \times 100\% = 49.6\%$

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

$CCFP_{10} = [1 - ((1.175E\text{-}05 + 2.59E\text{-}06)/ 2.88E\text{-}05)] \times 100\% = 50.2\%$

For the 15-year interval:

$CCFP_{15} = [1 - ((1.03E\text{-}05 + 3.89E\text{-}06)/ 2.88E\text{-}05)] \times 100\% = 50.7\%$

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

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

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

$\Delta CCFP_{\%} = CCFP_{10} - CCFP_{Base} = 0.63\%$

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

$\Delta CCFP_{\%} = CCFP_{15} - CCFP_{Base} = 1.1\%$

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

Non-Inspected Linear Surface

An alternative approach to show that the change in LERF meets the RG 1.174 acceptance guideline is to multiply the non-inspected area of the containment by the delta LERF from the 3-in-10 year interval to the 1-in-15 year interval.

The delta LERF is for Class 3b accidents from Tables 1 and 5 only:

$\Delta LERF = 3.89 \times 10^{-7} - 7.78 \times 10^{-8} = 3.11 \times 10^{-7}$

The non-inspected fraction has been calculated using dimensions from Reference 18 (see Attachment B).

Total area of liner = 104,321 ft^2

Non-inspected Area

Total Non-inspected area = 12,824 ft^2

% Not Inspected = (12,824/104,321) x100 = 12.3%

To account for additional containment liner surfaces that are not accessible inside containment the total non-inspected surface is rounded up to 13%.

The resulting change in LERF is calculated to be 0.13 x (3.11×10^{-7}) = 4.04×10^{-8}

Thus it has been independently shown that the change in LERF due to a 15-year ILRT meets the screening criterion in Reg. Guide 1.174.

External Event Sensitivity Analysis

The Millstone Unit 3 IPE (Ref. 5) has some limited discussion pertaining to external events. Table 1.4-2 in Ref. 5 shows percent contributions from internal, fire and seismic events for all release category frequencies. It is noteworthy to mention that for release category M4 which is the containment isolation failure, has a 89.9% contribution due to seismic, a 0.8% due to fire and 9.4% contribution due to internal events. The internal events frequency for release category M4 is reported as zero in Table A-1 in Attachment A. It appears that external events would have the largest impact on the EPRI class 7 event for the ILRT evaluation.

For the severe accident mitigation alternatives analysis (SAMA) for the MPS3 license renewal (Ref. 23) a factor was used to account for the potential impact of external events. The benefits of each SAMA were multiplied by a factor of 1.6 to account for the external events. This factor could be applied to the CDF used here to calculate the EPRI class 3a and 3b frequencies. Since class 3b represents a LERF then this multiplier would have the following effect on the ILRT analysis.

Baseline Class 3b frequency = 7.78E-08 x 1.6 = 1.24E-07/yr

15 year Class 3b frequency = 3.89E-07 x 1.6 = 6.22E-07/yr

The external events Δ LERF from Base to the15-year test is 4.98E-07/yr.

This compares to the internal events Base to 15-year Δ LERF as 3.11 E-07/yr.

Thus, it has been independently shown that with external events included the change in LERF due to a 15-year ILRT still meets the screening criterion in Reg. Guide 1.174. Since guidance in Reg. Guide 1.174 defines small changes in LERF in the range of 10^{-7}/yr, increasing the ILRT interval to 15 years is considered acceptable.

Liner Corrosion Analysis

The approach documented in the Calvert Cliffs Nuclear Power Plant submittal in Reference 19 was used to determine the change in likelihood, due to extending the ILRT, of detecting

liner corrosion. This likelihood was then used to determine the resulting change in risk. The following issues are addressed:

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

Assumptions

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

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

C. The liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the liner ages. Sensitivity studies are included that address doubling 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 Reference 19 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 of Reference 19 for Calvert Cliffs 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, assumed failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year was set to the historical failure rate. (See Table 5 from Ref. 19 for an example.)	<u>Year</u> <u>Failure Rate</u> 1 2.1E-3 avg 5-10 5.2E-3 15 1.4E-2 **15-year avg = 6.27E-3**	<u>Year</u> <u>Failure Rate</u> 1 5.0E-4 avg 5-10 1.3E-3 15 3.5E-3 **15-year avg = 1.57E-3**
3	**Increase in Flaw Likelihood Between 3 and 15 years** Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years. (See Tables 5 and 6 in Ref. 19.)	**8.7%**	**2.2%**
4	**Likelihood of Breach in Containment given Liner Flaw** The upper end pressure is consistent with the Calvert Cliffs Probabilistic Risk Assessment (PRA) Level 2 analysis. 0.1% is assumed for the lower end. Intermediate failure likelihoods are determined through logarithmic interpolation. The basemat is assumed to be 1/10 of the cylinder/dome analysis. The same value will be used for	Pressure (psia) Likelihood of Breach 20 0.1% 64.7 (ILRT) **1.1%** 100 7.02% 120 20.3% 150 100%	Pressure (psia) Likelihood of Breach 20 0.01% 64.7 (ILRT) **0.11%** 100 0.7% 120 2.0% 150 10%

Step	Description	Containment Cylinder and Dome	Containment Basemat
	MPS3 as was used for CCNP, since the containment design is somewhat similar. The design pressure of MPS3 is 45 psig versus 50 psig for CCNPP.		
5	**Visual Inspection Detection Failure Likelihood**	**10%** 5% failure to identity visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT). All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3*4*5)	**0.0096%** 8.7%*1.1%*10%	**0.0024%** 2.2%*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 = 0.0096% + 0.0024% = 0.012%.

The non-large early release frequency (LERF) containment over-pressurization failures for MPS3 is estimated at 1.31E-5 per year. This is based on the total CDF minus the Class 1, 3B and 8 frequencies from Table 1 (1.31E-5 = 2.88E-5 – (1.37E-05 + 7.78E-08 + 1.22E-06)). The total CDF for MPS3 is 2.88E-5. 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) = 0.00012 * 1.31E-5 = 1.57E-9 per year.

Thus, it has been independently shown that the increase in LERF, due to a liner corrosion failure, is **1.57E-9 per year** which meets the screening criterion of less than E-7 in Reg. Guide 1.174.

Results Summary

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 Dose Person-Rem**	**Base (3/10) Freq /Rx-yr**	**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	1.46E-05	1.65E+04	1.37E-05	2.27E-01	1.18E-05	1.94E-01		1.03E-05	1.70E-01	
2	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+0 0	0.00E+00		0.00E+00	0.00E+00	
3a	N/A	1.65E+05	7.78E-07	1.28E-01	2.59E-06	4.27E-01		3.89E-06	6.42E-01	
3b	N/A	5.78E+05	7.78E-08	4.49E-02	2.59E-07	1.50E-01		3.89E-07	2.25E-01	
4	N/A	0.00E+00	0.00E+00	0.00E+00	N/A	0.00E+00		N/A	0.00E+00	
5	N/A	0.00E+00	0.00E+00	0.00E+00	N/A	0.00E+00		N/A	0.00E+00	
6	N/A	0.00E+00	0.00E+00	0.00E+00	N/A	0.00E+00		N/A	0.00E+00	
7	1.30E-05	5.83E+05	1.30E-05	7.57E+00	1.30E-05	7.57E+00		1.30E-05	7.57E+00	
8	1.22E-06	4.10E+06	1.22E-06	5.00E+00	1.22E-06	5.00E+00		1.22E-06	5.00E+00	
	2.88E-05		**2.88E-05**	**12.98**	**2.88E-05**	**13.35**		**2.88E-05**	**13.61**	
	(CDF)		(CDF)		(CDF)			(CDF)		
ΔDR Change from Base							0.37			0.64
ILRT DR % of total dose				1.3			4.3			6.4
Δ% Change in ILRT DR from Base							3.0			5.1
LERF				3.2E-7			5.0E-7			6.3E-7
Δ LERF from base							1.8E-7			3.1E-7
CCFP, %				49.6			50.2			50.7
Δ CCFP, %							0.63			1.1
CDF	**2.88E-05**									

Identified Conservatisms in the ILRT Analysis

The results in Table 8 have shown that the 10 and 15 year metrics for some parameters are in line with expectations and others that do not meet expectations. As a result, it is important to analyze the results and to identify some conservatisms that exist in the methodology adopted for this ILRT extension analysis. These conservatisms include:

- A first conservatism exists in the conservatively high CDF that was used in this analysis. If the updated PRA model CDF (which is less) was used in this analysis, the resulting Class 3a and 3b frequencies would decrease the 10 and 15 year metrics proportionately.

- A second conservatism exists in the conservatively high baseline LERF that was used in this analysis. If the updated PRA model LERF (which is less) was used in this analysis, the baseline LERF would decrease the 10 and 15 year metrics proportionately.

7. Design Review

The most current drawings and procedures were used. This calculation does not perform a design verification or affect the design of the plant.

8. Enclosure

Enclosure 1-A: MPS3 Frequency and Dose Data

ENCLOSURE 1-A

MPS3 FREQUENCY AND DOSE DATA

Table A-1
MPS3 Frequency and Dose Data

Release Category	Frequency* Per year	Person-Rem**	EPRI Class	Description
M-1A	2.21E-07	1.01E+07	8	Cont Bypass, V-Sequence
M-1B	1.00E-06	2.77E+06	8	Cont Bypass, SGTR
M-2	0.00E+00	1.72E+06	7	Early Cont Failure Early Melt, No Sprays
M-3	0.00E+00	2.14E+06	7	Early Cont Failure Late Melt, No Sprays
M-4	0.00E+00	2.67E+06	2	Cont Iso Failure
M-5	1.96E-07	2.17E+06	7	Intermediate Cont Fail Late Melt, No Sprays
M-6	1.96E-07	2.55E+06	7	Intermediate Cont Fail Early Melt, No Sprays
M-7	7.79E-06	6.52E+04	7	Late Cont Fail No Sprays
M-8	0.00E+00	1.08E+06	7	Intermediate Cont Fail With Sprays
M-9	1.60E-06	9.49E+05	7	Late Cont Fail With Sprays
M-10	1.60E-06	2.04E+04	7	Basemat Failure No Sprays
M-11	1.60E-06	1.14E+04	7	Basemat Failure With Sprays
M-12	1.46E-05	1.65E+04	1	No Cont Failure
CDF	**2.88E-05**			

* Ref. 17
** Ref. 6

ATTACHMENT 3

<u>PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE</u>
<u>FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL</u>
<u>(LBDCR 06-MP3-010)</u>

<u>MARK-UP OF TECHNICAL SPECIFICATIONS</u>

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

PROCEDURES AND PROGRAMS (Continued)

2) Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

f. Containment Leakage Rate Testing Program

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

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 38.57 psig.

The maximum allowable containment leakage rate L_a, at P_a, shall be 0.3 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.042\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $<0.75\ L_a$ for Type A tests;

2) Air lock testing acceptance criteria are:

   a. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, seal leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Performanced Based Option of 10 CFR Part 50 Appendix J": The first Type A test performed after the January 6, 1998 Type A test shall be performed no later than January 6, 2013.

* An exemption to Appendix J, Option A, paragraph III.D.2(b)(ii), of 10 CFR Part 50, as approved by the NRC on December 6, 1985.

ATTACHMENT 4

**PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE
FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL
(LBDCR 06-MP3-010)**

RETYPED TECHNICAL SPECIFICATIONS PAGE

**DOMINION NUCLEAR CONNECTICUT, INC
MILLSTONE POWER STATION UNIT 3**

PROCEDURES AND PROGRAMS (Continued)

2) Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

f. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions*. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Performance Based Option of 10 CFR Part 50 Appendix J": The first Type A test performed after the January 6, 1998 Type A test shall be performed no later than January 6, 2013.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 38.57 psig.

The maximum allowable containment leakage rate L_a, at P_a, shall be 0.3 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq\ 0.042\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $< 0.75\ L_a$ for Type A tests;

2) Air lock testing acceptance criteria are:

   a. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, seal leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

* An exemption to Appendix J, Option A, paragraph III.D.2(b)(ii), of 10 CFR Part 50, as approved by the NRC on December 6, 1985.