Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
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Dominion

July 6, 2004

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

Serial No. 04-003
NL&OS/PRW R0
Docket No. 50-336
License No. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC. (DNC)
MILLSTONE POWER STATION UNIT 2
PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE
FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) requests an amendment to Facility Operating License Number DPR-65 in the form of a change to the Technical Specifications for Millstone Power Station Unit 2. 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.

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, respectively. A mark-up of Technical Specifications page 6-26 incorporating the proposed change to Technical Specification 6.19, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM," is provided in Attachment 3. The retyped page 6-26 is provided in Attachment 4.

The proposed change has been reviewed and approved by the Site Operations Review Committee and the Management Safety Review Committee. In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards. In addition, the proposed change has been determined to qualify for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). The basis for these determinations is included in Attachment 1.

To permit effective outage planning, it is requested that the NRC approve the proposed Technical Specification changes by March 1, 2005. Once approved the amendment will be implemented within 30 days.

There are no regulatory commitments contained within this letter.

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. Discussion of Change
2. Risk Assessment
3. Mark-up of Technical Specifications
4. Proposed Technical Specifications

cc: U.S. Nuclear Regulatory Commission
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

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 6th day of July, 2004.

My Commission Expires: 3/31/08.

ATTACHMENT 1

PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL

DISCUSSION OF CHANGE

MILLSTONE POWER STATION, UNIT 2
DOMINION NUCLEAR CONNECTICUT, INC.

DISCUSSION OF CHANGE

INTRODUCTION

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 2 Technical Specifications for the containment structure. A mark-up of Technical Specifications page 6-26 incorporating the proposed change to Technical Specification 6.19, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM," is provided in Attachment 3 and the retyped page 6-26 is provided in Attachment 4. The wording in Millstone Power Station Unit 2 Technical Specification 4.6.1.2 will remain the same. The proposed change will permit a one-time, five-year exception for Millstone Power Station Unit 2 from 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.

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.

BACKGROUND

The Millstone Power Station Unit 2 current ten-year Type A test interval ends on June 10, 2005. In order to meet the interval requirements of NEI 94-01, this test must be performed during either the Spring 2005 refueling outage, or using the fifteen-month extension provision, during the Fall 2006 refueling outage.

DNC is aware of an ongoing industry/NRC initiative to modify the existing performance-based leakage testing guidance to extend the maximum Type A test interval. Therefore, the requested exception is limited to five years for Millstone Unit 2, which is considered an adequate amount of time to complete the guidance change initiative.

DESCRIPTION OF CHANGE

This application for amendment to the Millstone Power Station Unit 2 Containment Leakage Rate Testing Program proposes to revise the technical specification surveillance requirement referenced in Section 4.6.1.2, Containment Leakage Rate Requirements. The proposed change will permit a one-time, five-year exception for Millstone Power Station Unit 2 from 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. The exception is to allow ILRT testing within fifteen years from the last ILRT, performed on June 10, 1995. This application represents a cost beneficial licensing change. The integrated leak rate test imposes significant expense on the station while the safety benefit of performing it within ten years, versus fifteen years, is minimal as described in the attached risk assessment. A mark-up of Technical Specifications page 6-26 incorporating the proposed change to Technical Specification 6.19, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM," is provided in Attachment 3 and the retyped page 6-26 is provided in Attachment 4. The wording in Millstone Unit 2 Technical Specification 4.6.1.2 will remain the same.

SAFETY IMPLICATIONS OF THE PROPOSED CHANGE

Implementing 10 CFR 50, Appendix J, Option B:

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

10 CFR 50, Appendix J, was revised, effective October 26,1995, to allow licensees to choose containment leakage testing under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements." Amendment 203 (Reference 2) was issued to Millstone Power Station Unit 2 to permit implementation of 10 CFR 50, Appendix J, Option B. Amendment 203 modified Technical Specification Section 4.6.1 and Technical Specification 6.19 which require testing in accordance with the Containment Leakage Testing Program and Regulatory Guide (RG) 1.163 (Reference 3) respectively. Regulatory Guide 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8–1994 (Reference 4), subject to several regulatory positions in the guide.

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

The 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 Type A test frequency did not result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not result in an increase in containment leakage.

The allowed frequency for testing was based upon a generic evaluation documented in NUREG-1493 (Reference 5). NUREG-1493 made the following observations with regard to decreasing 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."

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 December 1992 and June 1995 ILRTs, the current interval for Millstone Unit 2 is once every ten years.

Plant Specific Risk Assessment for the Extended ILRT Test Interval:

A risk assessment was performed in accordance with the guidelines set forth in NEI 94-01, the methodology used in EPRI TR-104285, 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. In addition, the results and findings from the Millstone Unit 2 Individual Plant Examination (IPE) and subsequent revised model were used in this risk assessment.

Method of Analysis:

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

The 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 and NUREG-1493. Namely, the analysis performed examined Millstone Power Station Unit 2 IPE 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.

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 0.83 x 10^{-8}/yr, based on internal events. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from ten to fifteen years is, therefore, considered non-risk significant. The calculation is included as an attachment to this technical specification change request. Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The DNC risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is estimated to be about 7.5 x 10^{-9} per year.

- The one-time change to the Type A test interval from ten years to fifteen increases the risk of those associated specific accident sequences by 4.32 x 10^{-5}%. In addition, the risk impact on the total integrated (fifteen-year increase) plant risk above baseline, for those accident sequences influenced by Type A testing is only 1.29 x 10^{-4}%. Therefore, the risk impact when compared to other severe accident risks is negligible.

Precedence

The proposed changes discussed within this license amendment request are similar to license amendments issued to 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.

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 2 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 Millstone Unit 2:

Date	As Found Leakage(*) WT%/day	Acceptance Limit(**) WT%/day	Test Pressure (psig)
December 12, 1992	0.2809	0.5	54.0
June 10, 1995	0.2559	0.5	54.0

* This is the leakage attributable to containment leakage as well as a number of Type B and Type C leakage components being tested as part of the Type A test.

** The leakage rate acceptance criteria are <0.60 L_a for the combined Type B and Type C tests, and <0.75 L_a (L_a = 0.5% of primary containment air by weight per day and is the leakage assumed in dose consequences) for Type A tests.

Plant Operational Performance:

During power operation control room instrumentation provides constant indication of containment pressure. If pressure rises, an alarm annunciates advising conditions are approaching the limits allowed by the Technical Specifications. This monitoring of the containment pressure equates to continuous on-line monitoring of the containment leakage during operation.

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 8, 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 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 2. 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 the relief request for Millstone Power Station Unit 2. 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 May 2000 and November 2003 refueling outages, DNC performed an IWE General Visual examination of the Containment Metal Liner (IWE - MC component). All accessible areas were examined. Some localized rust and surface anomalies were detected, most associated with blistering of the liner coating. Repairs meeting Code requirements were made during the past two refueling outages to eliminate most E-C classified items. Based on the current inspections and the associated engineering evaluations to date, DNC has not classified any areas as Examination Category E-C (accelerated degradation).

- 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 perform these inspections. 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.

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.

Fuel Transfer Tube Bellows:

There is one bellows installed in the Millstone Unit 2 containment on the outer tube of the fuel transfer tube containment penetration. The bellows compensates for differential motion and does not form the containment boundary. The containment boundary is the welded connection at the containment liner to the inner and outer tubes and the double O-ring blank flange on the inner (fuel transfer) tube. The blank flange is Type B tested each refueling outage and the welded connection is tested during the integrated leak rate test (ILRT). A manual isolation valve isolates the inner (fuel transfer) tube from the spent fuel pool in the spent fuel building.

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

The proposed revision to Technical Specifications permits a one-time extension to the current interval for Type A testing. The current test interval of ten years, which is based on the standard of good past performance, would be extended on a one-time basis to fifteen years from the last Type A test for Millstone Unit 2. In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based upon the following information:

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

   The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since extension of the containment Type A testing is not a physical plant modification that could alter the probability of accident occurrence nor, is 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 Unit 2 Type A test interval will not represent a significant increase in the consequences of an accident.

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

   The proposed revision to Technical Specifications adds a one-time extension to the current interval for Type A testing for Millstone Unit 2. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing does not create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure.

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

The proposed revision to Millstone Unit 2 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 Unit 2. 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 0.83×10^{-8}/yr, based on internal events. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, 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.

EVALUATION OF CATEGORICAL EXCLUSION FROM AN ENVIRONMENTAL ASSESSMENT

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

(i) The amendment involves no significant hazards consideration.

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

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

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

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

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

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

CONCLUSION

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

The Site Operations Review Committee (SORC) and the Management Safety Review Committee (MSRC) have reviewed this proposed change to the Technical Specifications and have concluded that it does not involve a significant hazards consideration and will not endanger the health and safety of the public.

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 2 Issuing Technical Specification Amendment 203, dated September 20,1996 to implement the requirements of 10 CFR 50, Appendix J, Option B for performance-based primary reactor containment leakage testing.

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

4. American National Standard ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements."

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

ATTACHMENT 2

PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE
FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL

RISK IMPACT ASSESSMENT

MILLSTONE POWER STATION, UNIT 2
DOMINION NUCLEAR CONNECTICUT, INC.

Approved 11/17/03 Effective 11/21/03

CALCULATION TITLE PAGE

Total Number of Pages: 31

Risk Impact Assessment of Extending Containment Type A Test Interval at Millstone Unit 2

TITLE

PRA03NQA-04057S2	0
CALCULATION No.	**Revision No.**
N/A	N/A
VENDOR CALCULATION No.	**Revision No.**

N/A

VENDOR NAME

NUCLEAR INDICATOR:	50.59 Evaluation or Screen Attached	**Calc. Supports DCR/MMOD/EE?**	**Calc. Supports Other Process?**
☐ **CAT1** ☐ **RWQA** ☐ **SBOQA** ☐ **FPQA** ☐ **ATWSQA** ☒ **NON-QA**	☐ YES ☒ NO	☐ YES ☒ NO	☐ YES ☒ NO

INCORPORATES:

CCN NO:	AGAINST REV.
N/A	N/A

N/A	N/A
Ref. No.	**Reference**

Executive Summary

A risk impact assessment of extending the plant's integrated leak rate test (ILRT) interval from ten to fifteen years has been made. The risk assessment is performed in accordance with the guidelines set forth in NEI 94-01 [1], the methodology used in EPRI TR-104285 [2] 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].

Approvals (Print & Sign Name)		
Preparer: M. G. Matras		Date:
Interdiscipline Reviewer:	Discipline:	Date:
Interdiscipline Reviewer :	Discipline:	Date:
Independent Reviewer: J. D. Leary		Date:
Engineering Approver: D. M. Bucheit		Date:

Installation Verification

☐ Calculation represents the installed configuration and approved licensing condition (Calculation of Record)

☒ N/A does not affect plant configuration (e.g., study, hypothetical analysis, etc.)

Preparer/Designer Engineer: (Print and Sign) M. G. Matras Date:

PassPort DATABASE INPUTs

Calculation Number: PRA03NQA-04057S2 Revision: 0

Vendor Calculation Number/Other: N/A Revision: N/A

CCN # N/A Calc. Voided: ☐ Yes ☒ No

Superseded By: N/A Supersedes Calc: N/A

Discipline (Up to 10) W

Unit (M1, M2, M3)	Project Reference (EWA, DCR or MMOD)	Component Id	Computer Code	Rev. No./ Level No.
N/A	N/A	N/A	N/A	N/A

PMMS CODES*					
Structure	System	Component	Reference Calculation	Rev. No.	CCN
N/A	N/A	N/A	PRA02NQA-03107S2	0	N/A
N/A	N/A	N/A	PRA99YQA-02863S2	3	N/A
N/A	N/A	N/A	PRA02NQA-03131S2	1	N/A

*The codes required must be alpha codes designed for structure, system and component.

NOTE: Avoid multiple item references on a line, e.g., LT 1210 A-D requires four separate lines.

Reference Drawing	Sheet	Rev. No.
N/A	N/A	N/A

Comments:

N/A

Referenced By Calculation	Impact Y	Impact N	AR Reference/Calc Change Ref.
N/A	N/A	N/A	N/A

PassPort DATABASE INPUTs

(Continuation Sheet)

Calculation Number: PRA03NQA-04057S2 Revision: 0

Vendor Calculation Number/Other: N/A Revision: N/A

CCN # N/A

Component ID	Reference Calculation	Rev No	CCN
N/A	N/A	N/A	N/A

*The codes required must be alpha codes designed for structure, system and component.

Reference Drawing	Sheet	Rev. No.
N/A	N/A	N/A

Comments:
N/A

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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 methodology used in EPRI TR-104285 [2] 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].

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 2 Nuclear Power Station (MPS2) 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 $1L_a$).

Millstone Unit 2 current ten-year Type A test is due to be performed during refueling outage (2R16), scheduled for April, 2005. Unit 2 Type A test was originally scheduled for April 2005 but will use the 5 year extension for the next scheduled test to be performed in 2010.

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 in NEI 94-01 [1], the methodology used in EPRI TR-104285 [2] 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] and 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, SP21208, “Containment Integrated Leak Rate Test, Type A”, Revision 0, 1995.
5) RF-Report, “Millstone Unit No 2 Individual Plant Examination For Severe Accident Vulnerabilities”, December 1993.
6) RF-Calc., PRA02NQA-03107S2 Revision 0, “MACCS2 model for Millstone Unit 2 Level 3 Application”, May 2003.
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., PRA99YQA-02863S2, Revision 3, “MP2 Final Quantification”, October 2002.
11) RF-Report., ERG-25203-ER-03-0028, MP2 Containment Risk Significant Valve Review, 7-11-03.
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'Connnor, "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-03131S2, Revision 1, “MP2 SAMA Impact on Containment Release Frequencies”, September 2003.
18) RF-Calc., M2-EV-04-0008 Revision 0, Technical Evaluation for MP2 Containment Liner Area, Millstone Unit 2, 03-01-04.
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-Calc., ERIN Report No. P0467020022-2011, Risk Assessment For TMI Unit 1 To Support ILRT (TYPE A) Interval Extension Request, 8-07-02.

4. Assumptions

The MP2 leakage rate (L_a) acceptance criteria is defined as:
L_a = 0.5 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 0.6-inch in diameter are considered as small leak rate releases. Break openings of greater than 0.7-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, 6, 7 and 8 are not impacted by the ILRT Type A Test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A.

13. For Class 6 sequences dominated by misalignment of containment isolation valves following a test/maintenance evolution, a conservative screening value of 1.0E-03 will be used to evaluate this class.

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] and NUREG-1493 [7]. Namely, the analysis performed examined MPS2 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 TR104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or steam generator manway leakage. (EPRI TR-104285 Class 3 sequences). 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]

6. Body of Calculation

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

This step involves the review of the MPS2 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 MPS2. 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).

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 La, under Appendix J for that plant. The allowable leakage rates (La), are typically 0.1 weight percent of containment volume per day for PWRs and 0.5 weight percent per day for BWRs (all measured at Pa, calculated peak containment pressure related to the design basis accident). Changes to leak rate testing frequencies do not affect this classification.

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

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

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

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

Class 6 Containment isolation failures include those leak paths not identified by the LLRTs. The type of penetration failures considered under this class includes those covered in the plant test and maintenance requirement or verified by in service inspection and testing (ISVIST) program. This failure to isolate is not typically identified in LLRT. Changes in Appendix J LLRT test intervals do not impact this class of accidents.

Class 7 Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.

Class 8 Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in class 8. Changes in Appendix J testing requirements do not typically impact these accidents, particularly for PWRs.

The frequencies for the above eight classes are calculated below. The Class 3–6 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 (1 L_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].

To estimate the failure probability given that no failures have occurred, a conservative estimate is obtained from the 95th percentile of the X^2 distribution. In statistical theory, the X^2 distribution can be used for statistical testing, goodness-of-fit tests, and evaluating s-confidence [13]. The X^2 distribution is really a family of distributions, which range in shape from that of the exponential to that of the normal distribution. Each distribution is identified by the degrees of freedom, v. For time-truncated tests (versus failure-truncated tests), an estimate of the probability of a large leak using the X^2 distribution can be calculated as X^2_{95} (v = 2n+2)/2N, where n represents the number of large leaks and N represents the number of ILRTs performed to date. With no large leaks (n = 0) in 144 events (N = 144) and $X^2_{95}(2)$ = 5.99, the 95th percentile estimate of the probability of a large leak is calculated as 5.99/(2*144) = 0.021.

To calculate the probability that a liner leak will be small (Event CLASS-3A), use was made of the data presented in NUREG-1493 [7]. The data found in NUREG-1493 states that 144 ILRTs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of 1.0L_a. However, of these 23 'failures' only 4 were found by an ILRT, the others were found by Type B and C testing or errors in test alignments. Therefore, the number of failures considered for "small releases" are 4-of-144. Similar to the event CLASS-3B probability, the estimated failure probability for small release is found by using the X^2 distribution. The X^2 distribution is calculated by n=4 (number of small leaks) and N=144 (number of events) which yields a X^2 (10) =18.3070. Therefore, the 95th percentile estimate of the probability of a small leak is calculated as 18.3070/(2*144) = 0.064.

The respective frequencies per year are determined as follows:

$$\text{CLASS-3A-FREQUENCY} = \text{PROB}_{\text{class-3a}} * \text{CDF}$$
$$\text{CLASS-3B-FREQUENCY} = \text{PROB}_{\text{class-3b}} * \text{CDF}$$

Where:

$\text{PROB}_{\text{class-3a}}$ = probability of small pre-existing containment liner leakage
= 0.064

$\text{PROB}_{\text{class-3b}}$ = probability of large pre-existing containment liner leakage
= 0.021

CLASS-3A-FREQUENCY = 0.064 * 7.17 x 10^{-5} /year = 4.59 x 10^{-6} /year

CLASS-3B-FREQUENCY = 0.021 * 7.17 x 10^{-5} /year = 1.51 x 10^{-7} /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. A conservative screening value of 1.0E-03 will be used to evaluate this class.

The low failure probabilities are based on the need for multiple failures, the presence of automatic closure signals, and control room indication. Based on the purpose of this calculation, and the fact that this failure class is not impacted by Type A testing, no further evaluation is needed. This is consistent with the EPRI guidance. However, in order to maintain consistency with the previously approved methodology (i.e.-PROBclass6 > 0), a conservative screening value of 1.0E-03 will be used to evaluate this class.

$$\text{FREQclass6} = (\text{Screening Value}) \times \text{CDF}$$

Screening Value = 1.0×10^{-3} [Assumed Conservative Value]

CLASS-6-FREQUENCY = $1.0 \times 10^{-3} * 7.17 \times 10^{-5}$ /year = 7.17×10^{-8} /year

For this analysis the associated maximum containment leakage for this group is $35L_a$

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact. The frequency per year for these sequences is 1.08×10^{-5} /year (Attachment A, Table A-1). For this analysis the associated maximum containment leakage for this group is $1L_a$. The MPS2 IPE did not model Class 3 or Class 6 type failures, therefore this needs to be accounted for in the Class 1 accident class. Using Reference 16 methodology, the frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3a, Class 3b and Class 6 in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$$\text{CLASS-1-FREQ} = \text{FREQ}_{\text{Class-1}} - (\text{FREQ}_{\text{Class3a}} + \text{FREQ}_{\text{Class3b}} + \text{FREQ}_{\text{Class6}})$$

$$\text{CLASS-1-FREQ} = 1.08 \times 10^{-5} - (4.59 \times 10^{-6} + 1.51 \times 10^{-7} + 7.17 \times 10^{-8})$$

CLASS-1-FREQ =5.99 x 10^{-6} /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 = 5.84 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 = 2.47 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
Mean Containment Frequencies Measures - Given Accident Class

Class	Description	Frequency (per Rx-year)
1	No Containment Failure	5.99 x 10^{-6}
2	Large Containment Isolation Failures (Failure-to-close)	0.00 x 10^{+0}
3a	Small Isolation Failures (Type A test)	4.59 x 10^{-6}
3b	Large Isolation Failures (Type A test)	1.51 x 10^{-7}
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)	7.17 x 10^{-8}
7	Severe Accident Phenomena Induced Failure (Early and late Failures)	5.84 x 10^{-5}
8	Containment Bypassed (SGTR & V-Sequence)	2.47 x 10^{-6}
CDF	Core Damage All CET End states	7.17 x 10^{-5}

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 MPS2 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 Class 7 accidents is the sum of all Class 7 accidents, excluding bypass failures, having CDF greater than zero from Table A-1.

Class 1 = 2.27×10^1 person-rem
Class 2 = 1.32×10^5 person-rem
Class 3a = $2.27 \times 10^1 * 10\ L_a = 2.27 \times 10^2$ person-rem
Class 3b = $2.27 \times 10^1 * 35\ L_a = 7.95 \times 10^2$ person-rem
Class 4 = Not analyzed
Class 5 = Not analyzed
Class 6 = $2.27 \times 10^1 * 35\ L_a = 7.95 \times 10^2$ person-rem
Class 7 = 1.90×10^6 person-rem
Class 8 = 4.96×10^6 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 sum of 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
Person-Rem Measures - Given Accident Class

Class	Description	Person-Rem (50-Miles)
1	No Containment Failure	2.27×10^1
2	Large Containment Isolation Failures (Failure-to-close)	1.32×10^5
3a	Small Isolation Failures (Type A test)	2.27×10^2
3b	Large Isolation Failures (Type A test)	7.95×10^2
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)	7.95×10^2

Class	Description	Person-Rem (50-Miles)
7	Failure Induced by Phenomena (Early and Late Failures)	1.90×10^{6}
8	Containment Bypassed (SGTR & V-Sequence)	4.96×10^{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	5.99×10^{-6}	2.27×10^{1}	1.360E-04
2	Large Isolation Failures (Failure-to-close)	$0.00 \times 10^{+0}$	1.32×10^{5}	0.000E+00
3a	Small Isolation Failures (Type A test)	4.59×10^{-6}	2.27×10^{2}	1.042E-03
3b	Large Isolation Failures (Type A test)	1.51×10^{-7}	7.95×10^{2}	1.200E-04
4	Small isolation Failure-to-Seal (Type B test)	N/A	N/A	0.000E+00
5	Small isolation Failure-to-Seal (Type C test)	N/A	N/A	0.000E+00
6	Other Isolation Failures (e.g., Dependent Failures)	7.17×10^{-8}	7.95×10^{2}	5.700E-05
7	Failure Induced by Phenomena (Early and Late Failures)	5.84×10^{-5}	1.90×10^{6}	1.110E+02
8	Containment Bypassed (SGTR & V-Sequence)	2.47×10^{-6}	4.96×10^{6}	1.225E+01
Total	All CET End States	7.17×10^{-5}	N/A	**1.23213E+02**

Based on the above values, using the same methodology as Reference 16, the baseline percent risk contribution due to Type A testing is as follows:

$\%Risk_{BASE} = [(CLASS3a_{BASE} + CLASS3b_{BASE}) / Total_{BASE}] \times 100$

Where:

$CLASS3a_{BASE}$ = class 3a person-rem/year = 1.04×10^{-3} person-rem/year [Table 3]

$CLASS3b_{BASE}$ = class 3b person-rem/year = 1.20×10^{-4} person-rem/year [Table 3]

$Total_{BASE}$ = total person-rem/year for baseline interval = 123.2 person-rem/year [Table 3]

$$\%RiskBASE = [(1.04 \times 10^{-3} + 1.20 \times 10^{-4}) / 123.2] \times 100\ \%$$
$$RiskBASE = 0.000943\%$$

Therefore, the baseline percent risk contribution, due to Type A testing is 0.000943%.

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

According to NUREG-1493 [7], relaxing the Type A ILRT interval from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months. (The average time for undetection is calculated by multiplying the test interval by ½ and multiplying by 12 to convert from years" to "months"). If the test interval is extended to 1 in 15 years, the average time that a leak detectable only by an ILRT test goes undetected increases to 90 months (1/2* 15 * 12). Since ILRTs only detect about 3% of leaks (the rest are identified during LLRTs), the result for a 10-yr ILRT interval is a 10% increase in the overall probability of leakage. This value is determined by multiplying 3% and the ratio of the average time for undetection for the increased ILRT test interval (60 months) to the baseline average time for undetection of 18 months. For a 15-yr-test interval, the result is a 15% increase in the overall probability of leakage (i.e., 3 * 90/18).

Risk Impact due to 10-year Test Interval

As previously stated, Type A tests impact only Class 1 and Class 3 sequences. In addition the increased probability of not detecting excessive leakage has no impact on the frequency of occurrence for Class 1 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large liner opening remains the same, even though the probability of not detecting the liner opening increases). Thus, only the frequency of Class 3 sequences is impacted. Therefore, for Class 3 sequences, the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in probability of leakage of 1.1. (Recall that for a 10-year interval there is a 10% increase on the overall probability of leakage).

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:

$\text{CLASS-1-FREQ} = \text{FREQ}_{\text{Class-1}} - (\text{FREQ}_{\text{Class3a}} + \text{FREQ}_{\text{Class3b}})*0.1$

$\text{CLASS-1-FREQ} = 5.99 \times 10^{-6} - (4.59 \times 10^{-6} + 1.51 \times 10^{-7}) *0.1$

CLASS-1-FREQ = 5.52E-06 **/year**

Likewise the 10 year Class 3a and 3b frequencies are increased by 10% of their base values to preserve the total CDF.
Class 3a = $4.59 \times 10^{-6} *1.1$ = 5.05E-06
Class 3b = $1.51 \times 10^{-7} *1.1$ = 1.66E-07
The results of this calculation 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	5.52E-06	2.27E+01	1.252E-04
2	Large Isolation Failures (Failure-to-close)	0.00E+00	1.32E+05	0.000E+00
3a	Small Isolation Failures (Type A test)	5.05E-06	2.27E+02	1.146E-03
3b	Large Isolation Failures (Type A test)	1.66E-07	7.95E+02	1.320E-04
4	Small isolation Failure-to-Seal (Type B test)	Not Analyzed	N/A	0.000E+00
5	Small isolation Failure-to-Seal (Type C test)	Not Analyzed	N/A	0.000E+00
6	Other Isolation Failures (e.g., Dependent Failures)	7.17×10^{-8}	7.95×10^{2}	5.700E-05
7	Failure Induced by Phenomena (Early and Late Failures)	5.84E-05	1.90E+06	1.110E+02
8	Bypass (SGTR)	2.47E-06	4.96E+06	1.225E+01
CDF	All CET End States	**7.17E-05**	N/A	**1.23213E+02**

Based on the above values, the Type A 10-year test frequency percent risk contribution ($\%Risk_{10}$) for Class 3 is as follows:

$$\%Risk_{10} = [(CLASS3a_{10} + CLASS3b_{10}) / Total_{10}] \times 100$$

Where:

$CLASS3a_{10}$ = class 3a person-rem/year = 1.15×10^{-3} person-rem/year [Table 4]

$CLASS3b_{10}$ = class 3b person-rem/year = 1.32×10^{-4} person-rem/year [Table 4]

$Total_{10}$ = total person-rem year for 10-year interval = 123.2 person-rem/year [Table 4]

$$\%Risk_{10} = [(1.15 \times 10^{-3} + 1.32 \times 10^{-4}) / 123.2] \times 100$$
$$\%Risk_{10} = 0.001037\%$$

Therefore, the total 10-year test frequency ILRT interval percent risk contribution due to Type A testing is 0.001037%.

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

$$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE} \times 100.0$$

Where:

$Total_{BASE}$ =total person-rem/year for baseline interval = **1.23213E+02** person-rem/year [Table 3]

$Total_{10}$ = total person-rem/year for 10-year interval = **1.23213E+02** person-rem/year [Table 4]

$\Delta\%Risk_{10}$ = [(**1.23213E+02** - **1.23213E+02**) / **1.23213E+02**] x 100.0
$\Delta\%Risk_{10}$ = 8.57E-5%

Therefore, the person-rem/year increase in risk contribution because of relaxed ten-year ILRT test frequency from the baseline three-in-ten-years to 1-in-ten-years is approximately 8.57E-5%.

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. For this case the value is 15 percent or 1.15. (Recall that for a 10-year interval there is a 10% increase on the overall probability of leakage). In addition, the containment leakage used for the 10-year test interval for both Class 1 and Class 3 are used in the 15-year interval evaluation.

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 = $FREQ_{Class\text{-}1}$ - ($FREQ_{Class3a}$ + $FREQ_{Class3b}$)*0.15

CLASS-1-FREQ = 5.99×10^{-6} - ($4.59 \times 10^{-6} + 1.51 \times 10^{-7}$) *0.15

CLASS-1-FREQ =5.28×10^{-6} /year

Likewise the 15 year Class 3a and 3b frequencies are increased by 15% of their base values to preserve the total CDF.
Class 3a = 4.59×10^{-6} * 1.15 = 5.28E-06
Class 3b = 1.51×10^{-7} * 1.15 = 1.74E-07
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	5.28E-06	2.27×10^1	1.199E-04
2	Large Isolation Failures (Failure-to-close)	0.00E+00	1.32×10^5	0.000E+00
3a	Small Isolation Failures (Type A test)	5.28E-06	2.27×10^2	1.199E-03
3b	Large Isolation Failures (Type A test)	1.74E-07	7.95×10^2	1.383E-04
4	Small isolation Failure-to-Seal (Type B test)	N/A	N/A	0.000E+00

Class	Description	Frequency (per Rx-yr)	Person-Rem (50-Miles)	Person-Rem/yr (50-Miles)
5	Small isolation Failure-to-Seal (Type C test)	N/A	N/A	0.000E+00
6	Other Isolation Failures (e.g., Dependent Failures)	7.17×10^{-8}	7.95×10^{2}	5.700E-05
7	Failure Induced by Phenomena (Early and Late Failures)	5.84E-05	1.90×10^{6}	1.110E+02
8	Bypass (SGTR)	2.47E-06	4.96×10^{6}	1.225E+01
CDF	All CET End States	**7.17E-05**	N/A	**1.23213E+02**

Based on the above values, the Type A 15-year test frequency percent risk contribution ($\%Risk_{15}$) for Class 3 is as follows:

$$\%Risk_{15} = [(CLASS3a_{15} + CLASS3b_{15}) / Total_{15}] \times 100$$

Where:

$CLASS3a_{15}$ = class 3a person-rem/year = 1.199E-03 person-rem/year [Table 5]

$CLASS3b_{15}$ = class 3b person-rem/year = 1.383E-04 person-rem/year [Table 5]

$Total_{15}$ = total person-rem year for 15-year interval = 123.2 person-rem/year [Table 5]

$\%Risk_{15} = [(1.199E\text{-}03 + 1.383E\text{-}04) / 123.2] \times 100\ \%$
$\%Risk_{15} = 0.001085\%$
Therefore, the total Type A 15-year ILRT interval risk contribution of leakage, represented by Class 1 and Class 3 accident scenarios is 0.001085%.

<u>The percent increase on the total integrated plant risk due to a five-year increase over the 10 year ILRT is computed as follows:</u>

$$\%TOTAL_{10\text{-}15} = [(TOTAL_{15} - TOTAL_{10}) / TOTAL_{10}] \times 100$$

Where:
$TOTAL_{10}$ = total person-rem/year for 10-year interval = **1.23213E+02**person-rem/year [Table 4]
$TOTAL_{15}$ = total person-rem/year for 15-year interval = **1.23213E+02**person-rem/year [Table 5]

$\%TOTAL_{10\text{-}15} = [(\mathbf{1.23213E+02} - \mathbf{1.23213E+02}) / \mathbf{1.23213E+02}] \times 100$
$\%TOTAL_{10\text{-}15} = 4.32E\text{-}5\%$

Therefore, the person-rem/year increase in risk contribution from extending the ILRT test frequency from the current once-per-ten-year interval to once-per-fifteen years (a 5 year increase) is 4.32E-5%.

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

$\Delta\%Risk_{15} = [(TOTAL_{15} - Total_{BASE}) / Total_{BASE} \times 100$

Where:
$Total_{BASE}$ = total person-rem/year for baseline interval = **1.23213E+02** person-rem/year [Table 3]
$Total_{15}$ = total person-rem/year for 15-year interval = **1.23213E+02** person-rem/year [Table 5]

$\Delta\%Risk_{15} = [(1.23213E+02 - 1.23213E+02) / 1.23213E+02] \times 100.0$
$\Delta\%Risk_{15} = 1.289E-4\%$

Therefore, the person-rem/yr increase in risk contribution associated with relaxing the ILRT test frequency from the baseline three in ten years to once-per-fifteen years is approximately 1.289E-4%.

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

The one time extension of increasing the Type A test interval involves establishing the success criteria for a large release. 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.

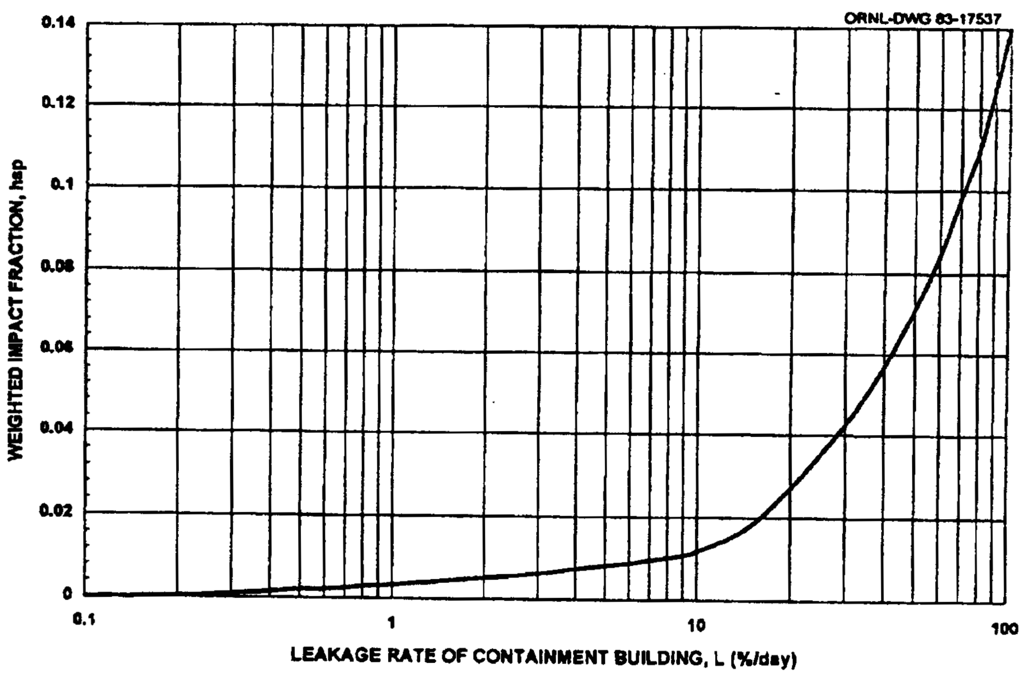
The containment leak size for the corresponding leak rate was calculated using the same methodology as in Reference 20. The effect of containment leak size on the containment leak rate is shown in Table 6. In addition, Oak Ridge National Laboratory (ORNL) [14] completed a study evaluating the impact 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 slightly larger than 0.6 inches. It is to be noted that for MPS2 a containment diameter of 0.6 inches was calculated as opposed to 2.0 inches for Indian Point 3. This difference in containment leak diameter is due to the difference in containment size between MPS2 and Indian Point 3. Therefore, this study defines small leakage as containment leakage resulting from an opening of 0.34 in^2 or less, large leakage as greater than 35%/day and negligible leakage as 0. 5% /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	0.005	0.5
0.102	0.010	1.0
0.324	0.097	10.0
0.606	0.341	35.0
1.024	0.975	100.0
1.996	3.703	380.0
5.016	23.39	2400.0

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.

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 MPS2 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 MPS2. This frequency, based on a ten-year test interval, is 1.66×10^{-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] defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

As described in Step 3, extending the ILRT interval from once-per-10 years to once-per-15 years will increase the average time that a leak detectable only by an ILRT goes undetected from 60 to 90 months (0.5*15*12). Since ILRTs only detect about 3% of leaks (the rest are identified during LLRTs), the result for a 15-yr ILRT interval is a 15% increase in the overall probability of leakage (3 * 90/18) versus 10% for a 10-yr ILRT interval. Thus, increasing the ILRT test interval from 10 years to 15 years results in a 5% increase in the overall probability of leakage. Multiplying the above LERF frequency (1.66×10^{-7}/yr) by the increase in overall probability of leakage (0.05) gives an increase in LERF of 0.83×10^{-8}/yr. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1.0 E-7/yr, increasing the ILRT interval to 15 years is non-risk significant.

It should be noted that if the risk increase is measured from the original 3-in-10-year interval, the increase in LERF is 1.51×10^{-7}/yr from Class 3B sequences (Table 3) multiplied by the 12% incremental increase in overall probability for a fifteen-year test interval (i.e., 15% - 3%) is 1.81×10^{-8}/yr, which is still below the 1.0E-7/yr screening criterion in Reg. Guide 1.174).

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. 1), 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).
CCFP percent for a given ILRT interval can be calculated using the following equation from Reference 21.

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

For the 10-year interval:

$CCFP_{10} = [1 - ((5.52E\text{-}06 + 5.05E\text{-}06)/7.17E\text{-}05)] \times 100\%$
$= 85.258\%$

For the 15-year interval:

$CCFP_{15} = [1 - ((5.28E\text{-}06 + 5.28E\text{-}06)/7.17E\text{-}05)] \times 100\%$
$= 85.272\%$

Therefore the change in the conditional containment failure probability is:

$\Delta CCFP_{\%} = CCFP_{15} - CCFP_{10} = 0.01\%$

This change in $CCFP_{\%}$ of less than 1 percentage point is insignificant 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:
$\Delta LERF = 1.74 \times 10^{-7} - 1.51 \times 10^{-7} = 0.23 \times 10^{-7}$

The non-inspected fraction has been calculated using dimensions from Reference 18.
Total area of liner = 84,456 ft^2

Non-inspected area
Total Non inspected area = 17,404 ft^2

% Not Inspected = (17,404/84,456) x100 = 20.6%

To account for additional containment liner surfaces that are not accessible inside containment the total non-inspected surface is rounded up to 25%.
The resulting change in LERF is calculated to be
$0.25 \times (0.23 \times 10^{-7}) = 5.8 \times 10^{-9}$

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.

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 increase 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. Calvert Cliffs sensitivity studies are considered appropriate for MP2 since they both have the same CE reactor supplier with similar reactor power level and AE containment design.

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 *5.5* 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.)	**Year** **Failure Rate** 1 2.1E-3 avg 5-10 5.2E-3 15 1.4E-2 **15 year avg = 6.27E-3**	**Year** **Failure Rate** 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 MPS2 as was used for CCNP, since the containment design is similar. See discussion on Table 4.4.1-1 in MPS2 IPE, Containment Back-End Analysis of Reference 5.	Pressure (psia) Likelihood of Breach 20 0.1% 68.7 (ILRT) **1.1%** 100 7.02% 120 20.3% 150 100%	Pressure (psia) Likelihood of Breach 20 0.01% 68.7 (ILRT) **0.11%** 100 0.7% 120 2.0% 150 10%

Step	Description	Containment Cylinder and Dome	Containment Basemat
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 MPS2 is estimated at 6.31E-5 per year. This is based on the total CDF minus the Class 1,3B and 8 frequencies from Table 1 (6.31E-5 = 7.17E-5 –(5.99E-6+1.51E-7 +2.47E-6)). The total CDF for MPS2 is 7.17E-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 * 6.31E-5 = 7.57E-9 per year.

Thus it has been independently shown that the increase in LERF due to a liner corrosion failure is **7.57E-9 per year** which meets the screening criterion of less than E-7 in Reg. Guide 1.174.

Results Summary

1. The baseline (3 in ten years) percent risk contribution of Type A testing, represented by Class 3 accident scenarios is 0.000943%.

2. Type A 10-year ILRT interval percent risk contribution of Type A testing, represented by Class 3 accident scenarios is 0.001037%.

3. Type A 15-year ILRT interval percent risk contribution of leakage, represented by Class 3 accident scenarios is 0.001085%.

4. The person-rem/year ($\Delta\%Risk_5$) increase in risk contribution from extending the ILRT test frequency from the 1 in ten years interval, to once-per-15 years (a 5 year increase) is 4.32E-5%.

5. The person-rem/year ($\Delta\%Risk_{10}$) increase in risk contribution from extending the ILRT test frequency from the baseline 3 in ten years interval to once-per-10 years is 8.57E-5%.

6. The person-rem/year ($\Delta\%Risk_{15}$) increase in risk contribution from extending the ILRT test frequency from the baseline case, 3 in ten years interval, to once-per-15 years is 1.29 E-4%.

7. The increase in LERF from the baseline 3-in-10-year interval, to once-per-15 years is 1.81 x 10^{-8}/yr which is below the 1.0E-7/yr screening criterion.

8. The change in the conditional containment failure probability is 0.01% which is considered insignificant from a risk perspective.

9. Other salient results are summarized in Table 8.

Table 8
Summary of Risk Impact on Extending Type A ILRT Test Frequency

Class*	Risk Impact (BASE)**	Risk Impact (10-years)***	Risk Impact (15-years)****
3a and 3b (% of integrated value based on $10L_a$ Class 3a and $35L_a$, for Class 3b)	0.000943%	0.001037%	0.001085%
Total Integrated Risk (person-rem/year)	123.2	123.2	123.2

* Only accident sequences impacted by a change in Type A test frequency are evaluated. These are sequences 3A and 3B
** MPS2 Revised IPE baseline values
***Type A ILRT test interval of 1-in-10-years
****Type A ILRT test interval of 1-in-15-years

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. Attachment

Attachment A: MP2 Frequency and Dose Data

Attachment A:
MPS2 Frequency and Dose Data

Table A-1
MPS2 Frequency and Dose Data

Release Category	Frequency* Per year	Person-Rem**	EPRI Class	Description
M-1A	1.07E-07	3.90E+06	8	Cont Bypass, V-Sequence
M-1B	2.36E-06	1.06E+06	8	Cont Bypass, SGTR
M-2	0.00E+00	1.18E+05	7	Early Cont Failure Early melt, No Sprays
M-3	6.86E-07	7.04E+05	7	Early Cont Failure Late melt, No Sprays
M-4	0.00E+00	1.32E+05	2	Cont Iso Failure
M-5	5.48E-06	4.47E+05	7	Intermediate Cont Fail Late melt, No Sprays
M-6	1.37E-05	4.47E+05	7	Intermediate Cont Fail Early melt, No Sprays
M-7	2.14E-05	7.62E+04	7	Late Cont Fail No sprays
M-8	1.71E-05	2.23E+05	7	Intermediate Cont Fail With Sprays
M-9	0.00E+00	9.34E+00	7	Late Cont Fail With sprays
M-10	0.00E+00	1.02E+06	7	Basemat Failure No sprays
M-11	0.00E+00	1.56E+01	7	Basemat Failure With sprays
M-12	1.08E-05	2.27E+01	1	No Cont Failure
CDF	**7.17E-05**			

* Ref. 17
** Ref. 6

ATTACHMENT 3

PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL

MARK-UP OF TECHNICAL SPECIFICATIONS

MILLSTONE POWER STATION, UNIT 2
DOMINION NUCLEAR CONNECTICUT, INC.

6.19 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the primary containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995~~.~~, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Performance-Based Option of 10 CFR Part 50, Appendix J,"

The peak calculated primary Containment internal pressure for the design basis loss of coolant accident is P_a.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.5% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $< 1.0\ L_a$. 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 $< 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

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

   2. For each door, pressure decay is ≤ 0.1 psig when pressurized to ≥ 25 psig for at least 15 minutes.

The provisions of SR 4.0.2 do not apply for test frequencies specified in the Primary Containment Leakage Rate Testing Program.

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

6.20 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the REMODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the REMODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10CFR 20.1001-20.2402;

ATTACHMENT 4

PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE
FIVE-YEAR EXTENSION OF TYPE A TEST INTERVAL

RETYPED TECHNICAL SPECIFICATIONS PAGE

MILLSTONE POWER STATION, UNIT 2
DOMINION NUCLEAR CONNECTICUT, INC

6.19 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the primary containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Performance-Based Option of 10 CFR Part 50, Appendix J."

The peak calculated primary Containment internal pressure for the design basis loss of coolant accident is P_a.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.5% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $< 1.0\ L_a$. 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 $< 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

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

2. For each door, pressure decay is ≤ 0.1 psig when pressurized to ≥ 25 psig for at least 15 minutes.

The provisions of SR 4.0.2 do not apply for test frequencies specified in the Primary Containment Leakage Rate Testing Program.

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

6.20 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the REMODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the REMODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10CFR 20.1001-20.2402;