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Constellation Energy
Nine Mile Point Nuclear Station

August 13, 2009

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 2; Docket No. 50-410

License Amendment Request Supplement Pursuant to 10 CFR 50.90: Adoption of NEI 94-01, Revision 2-A, and Extension of Primary Containment Integrated Leakage Rate Test Interval to Fifteen (15) Years - Technical Specification 5.5.12, 10 CFR 50 Appendix J Testing Program Plan

REFERENCE: (a) Letter from S. Belcher (NMPNS) to Document Control Desk (NRC), dated June 29, 2009, License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NEI 94-01, Revision 2, and Extension of Primary Containment Integrated Leakage Rate Test Interval to Fifteen (15) Years - Technical Specification 5.5.12, 10 CFR 50 Appendix J Testing Program Plan

Nine Mile Point Nuclear Station, LLC (NMPNS) hereby supplements the request for an amendment to the Nine Mile Point Unit 2 (NMP2) Renewed Facility Operating License NPF-69 that was submitted in Reference (a). The proposed change would revise NMP2 Technical Specification (TS) 5.5.12, "10 CFR 50 Appendix J Testing Program Plan," by replacing the reference to Regulatory Guide 1.163 with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 2-A, as the implementation document used by NMPNS to develop the NMP2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. This supplement addresses the concern forwarded to NMPNS by the NRC in an email dated August 5, 2009, by specifically referencing NEI 94-01, Revision 2-A, in the proposed TS change. Corresponding administrative revisions to the "Evaluation of the Proposed Change" and to the "Significant Hazards Consideration" portions of the original license amendment request have also been incorporated and are enclosed. The revisions are indicated by a vertical bar drawn in the right hand margin of affected pages.

A017
NRR

Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this license amendment request supplement, with Enclosure, to the appropriate state representative.

Should you have any questions regarding the information in this submittal, please contact T. F. Syrell, Licensing Director, at (315) 349-5219.

Very truly yours,

STATE OF NEW YORK :
: **TO WIT:**
COUNTY OF OSWEGO :

I, Sam Belcher, being duly sworn, state that I am Vice President-Nine Mile Point, and that I am duly authorized to execute and file this license amendment request supplement on behalf of Nine Mile Point Nuclear Station, LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Nine Mile Point employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 13 day of August, 2009.

WITNESS my Hand and Notarial Seal:

Lisa M. Clark
Notary Public

My Commission Expires:

9/12/09
Date

LISA M. CLARK
Notary Public in the State of New York
Oswego County Reg. No. 01CL6029220
My Commission Expires 9/12/09

SB/DEV

Enclosure: Evaluation of the Proposed Change

cc: S. J. Collins, NRC
R. V. Guzman, NRC
Resident Inspector, NRC
A. L. Peterson, NYSERDA

ENCLOSURE

EVALUATION OF THE PROPOSED CHANGE

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ATTACHMENTS

1. Proposed Technical Specification Change (Mark-up)
2. Risk Impact Assessment of Extending Containment Type "A" Test Interval for NMP2 (2NER-PR-003)

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Operating License NPF-69 for Nine Mile Point Unit 2 (NMP2).

The proposed amendment revises NMP2 Technical Specification (TS) 5.5.12, "10 CFR 50 Appendix J Testing Program Plan," by replacing the reference to Regulatory Guide (RG) 1.163 (Reference 1) with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 2-A, as the implementation document used by Nine Mile Point Nuclear Station, LLC (NMPNS) to develop the NMP2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J.

Revision 2-A of NEI 94-01 (Reference 4) describes an approach for implementing the optional performance-based requirements of Option B, including provisions for extending primary containment integrated leak rate test (ILRT) intervals to 15 years, and incorporates the regulatory positions stated in RG 1.163. In the safety evaluation (SE) issued by NRC letter dated June 25, 2008 (Reference 3), the NRC concluded that NEI 94-01, Revision 2 (Reference 2), describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the safety evaluation.

In accordance with the guidance in NEI 94-01, Revision 2-A (the NRC-accepted version of Revision 2), NMPNS proposes to extend the interval for the primary containment ILRT, which is required to be performed by 10 CFR 50, Appendix J, from 10 years to no longer than 15 years from the last ILRT. The current 10-year ILRT for NMP2 is due by April 11, 2010, which would require the test to be performed during the spring 2010 refueling outage. The proposed amendment would allow the next ILRT for NMP2 to be performed within 15 years from the last ILRT (i.e., by April 11, 2015), as opposed to the current 10-year interval, and would allow successive ILRTs to be performed at 15-year intervals (assuming acceptable performance history). The performance of fewer ILRTs will result in significant savings in radiation exposure to personnel, cost, and critical path time during future refueling outages.

The technical analysis for the proposed amendment is consistent with the guidance in Section 9.2.3 of NEI 94-01, Revision 2-A, including performance of a confirmatory plant-specific risk assessment, and addresses the limitations and conditions identified in the NRC safety evaluation (Reference 3).

2.0 DETAILED DESCRIPTION

2.1 Description of the Proposed Change

NMP2 TS 5.5.12, "10 CFR 50 Appendix J Testing Program Plan," Item a, currently states:

> *A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B with the exemptions stated in Section 2.D(ii) of the Operating License. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled, "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:*

1. *The measured leakage of main steam isolation valves (MSIVs) is excluded from the combined leakage rate of 0.6 L_a, and as-found testing is not required to be performed on the MSIVs.*

2. *Primary containment air lock door seals are tested prior to re-establishing primary containment OPERABILITY when something has been done that would bring into question the validity of the previous air lock door seal test.*

The proposed change would revise the initial paragraph of TS 5.5.12.a by replacing the reference to RG 1.163 with a reference to NEI 94-01, Revision 2-A (changes underlined), and would revise the first listed exception by deleting the portion regarding as-found testing of the MSIVs (marked with a strikethrough), as shown below:

A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B with the exemptions stated in Section 2.D(ii) of the Operating License. This program shall be in accordance with the guidelines contained in <u>NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008,</u> with the following exceptions:

1. *The measured leakage of main steam isolation valves (MSIVs) is excluded from the combined leakage rate of 0.6 L_a~~, and as-found testing is not required to be performed on the MSIVs~~.*

2. *Primary containment air lock door seals are tested prior to re-establishing primary containment OPERABILITY when something has been done that would bring into question the validity of the previous air lock door seal test.*

Attachment 2 to this Enclosure contains existing TS page 5.5-11 marked up to show the proposed changes to TS 5.5.12.a.

2.2 Background

2.2.1 <u>Description of Primary Containment System</u>

The primary containment is described in Updated Safety Analysis Report (USAR) Sections 3.8.1, 3.8.3, and 6.2.1. The Mark II pressure suppression containment system consists of the drywell, the pressure suppression chamber (which stores a large volume of water) and the drywell floor which separates the drywell and suppression chamber. The primary containment structure houses the reactor vessel, the reactor recirculation system, and other branch connections of the reactor coolant pressure boundary (RCPB).

The original design of the primary containment preceded the issuance of American Society of Mechanical Engineers (ASME) Section III, Division 2. As a result, the reinforced concrete primary containment was designed and constructed to the requirements of the American Concrete Institute (ACI), Building Code Requirements for Reinforced Concrete, ACI 318-71. The primary containment steel liner is designed following the requirements of ASME Section III, Division 1, and the regions around the containment penetrations are designed to meet the requirements of ASME Section III, Division 2.

The drywell is a steel-lined reinforced concrete vessel in the shape of a frustum of two cones, closed by a dome with a torispherical head. The drywell has a base diameter of approximately 91 ft and a top diameter of approximately 34 ft. The floor of the drywell serves both as a pressure barrier between the drywell and the suppression chamber and as the support structure for the reactor pedestal, downcomer vent lines, and other penetrations. The steel liner on the top surface of the drywell floor functions as a positive gas-tight membrane between the drywell and the suppression chamber to ensure that steam can enter the suppression chamber only through the downcomer vent lines or the safety/relief valve (SRV) discharge lines. This liner is anchored and seal-welded to the reactor pedestal wall, the primary containment liner knuckle, and the drywell floor penetrations and embedments. The drywell floor slopes away from the containment wall liner and towards the drywell floor drain system.

The drywell houses the reactor vessel and associated equipment. The primary function of the drywell is to contain the radioactivity and withstand pressures and temperatures resulting from a breach of the RCPB, up to and including an instantaneous circumferential break of a single reactor recirculation pump suction pipe, and to provide a holdup time for decay of any radioactive material released. The drywell is designed to resist the forces of an internal design pressure of 45 psig in combination with thermal, seismic, and other loads as outlined in USAR Chapter 3.

The pressure suppression chamber is a cylindrical, stainless steel clad, steel-lined, reinforced concrete vessel located below the drywell, having an inside diameter of approximately 91 feet. The foundation mat, to which the vessel is anchored, is lined with steel plates within the inside diameter of the cylinder. The steel plates are welded to each other and to steel embedments to maintain the primary containment function of a gas-tight enclosure.

The pressure suppression pool, which is contained within the pressure suppression chamber, stores sufficient water to condense the steam released from blowdown of the reactor coolant system after a loss of coolant accident (LOCA) or from SRV discharge during accident or normal operational transients. Steam is transferred to the pressure suppression pool by the downcomer vent lines and the discharge piping of the SRVs. In addition to serving as a heat sink for transients and accidents, the pressure suppression pool also provides a reservoir of water for the core standby cooling systems. The downcomer vent lines are open to the drywell and submerged below the low water level of the suppression pool, providing a path for uncondensed steam to enter the pool. The downcomers project 3 to 6 inches above the sloped drywell floor so that small quantities of water leakage flow past the downcomers and are collected in the drywell floor drain system.

Vacuum breakers provide a return flow path from the suppression chamber gas space to the drywell. The vacuum breakers are designed to limit the negative differential pressure between the drywell and the suppression chamber to less than the design value of 10 psid. The vacuum breaker valves are mounted in piping that connects the drywell and suppression chamber. Each of the four vacuum breaker flow paths has two relief valves in series to ensure a leak-tight boundary under positive drywell-to-suppression chamber differential pressure conditions. Since the vacuum breakers are located inside the drywell, they do not form an extension of the primary containment boundary.

2.2.2 Testing Requirements of 10 CFR 50, Appendix J

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS, and that periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and the systems and components penetrating primary containment. The limitation on containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident. Appendix J identifies three types of required tests: (1) Type A tests, intended to measure the primary containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for primary containment penetrations; and (3) Type C tests, intended to measure containment isolation valve leakage rates. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing.

In 1995, 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," was amended to provide a performance-based Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term "performance-based" in 10 CFR 50 Appendix J refers to both the performance history necessary to extend test intervals as well as to the criteria necessary to meet the requirements of Option B.

Also in 1995, RG 1.163 (Reference 1) was issued. The RG endorsed NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 5), with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency for the containment Type A (ILRT) test from three tests in 10 years to one test in 10 years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, "Performance-Based Containment Leak-Test Program" (Reference 6), and EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals" (Reference 7), both of which showed that the risk increase associated with extending the ILRT surveillance interval was very small.

By letter dated August 31, 2007, the NEI submitted Revision 2 of NEI 94-01 and EPRI TR-1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," to the NRC staff for review.

NEI 94-01, Revision 2 (Reference 2), describes an approach for implementing the optional performance-based requirements of Option B described in 10 CFR Part 50, Appendix J, which includes provisions for extending Type A (ILRT) intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. This method uses industry performance data, plant-specific performance data, and risk insights in determining the appropriate testing frequency. NEI 94-01, Revision 2, also discusses the performance factors that licensees must consider in determining test intervals. However, it does not address how to perform the tests because these details can be found in existing documents (e.g., ANSI/ANS-56.8-2002). The NRC final SE issued by letter dated June 25, 2008 (Reference 3), documents the NRC's evaluation and acceptance of NEI 94-01, Revision 2, subject to the specific limitations and conditions listed in Section 4.1 of the SE. The accepted version of NEI 94-01 has subsequently been issued as Revision 2-A dated October 2008 (Reference 4).

EPRI TR-1009325, Revision 2 (Reference 8), provides a risk impact assessment for optimized ILRT intervals of up to 15 years, utilizing current industry performance data and risk-informed guidance, primarily Revision 1 of RG 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 9). The NRC final SE issued by letter dated June 25, 2008, documents the NRC's evaluation and acceptance of EPRI TR-1009325, Revision 2, subject to the specific limitations and conditions listed in Section 4.2 of the SE. An accepted version of EPRI TR-1009325 has subsequently been issued as Revision 2-A (also identified as TR-1018243) dated October 2008 (Reference 10).

2.2.3 Current NMP2 TS Requirements

On August 13, 1996, the NRC approved License Amendment No. 74 for NMP2 (Reference 11), authorizing implementation of the containment leak rate testing requirements of 10 CFR 50, Appendix J, Option B, with exemptions stated in Section 2.D(ii) of the NMP2 Operating License. The amendment added TS 6.8.4.f, "10 CFR 50 Appendix J Testing Program Plan," to require Type A, B and C testing in accordance with RG 1.163, with the following two identified exceptions:

1. The measured leakage of main steam isolation valves (MSIVs) is excluded from the combined leakage rate of 0.6 L_a, and as-found testing is not required to be performed on the MSIVs.
2. Primary containment air lock door seals are tested prior to re-establishing primary containment OPERABILITY when something has been done that would bring into question the validity of the previous air lock door seal test.

The NRC approved these exceptions to RG 1.163 in the safety evaluation that accompanied issuance of License Amendment No. 74 (Reference 11).

TS 6.8.4.f was subsequently re-numbered as TS 5.5.12 in License Amendment No. 91 (conversion to improved TS), issued by NRC letter dated February 15, 2000 (Reference 12). As described in TS 5.5.12, the maximum allowable primary containment leakage rate (L_a) is 1.1% of primary containment air weight per day at the peak calculated design basis LOCA containment internal pressure (P_a) of 39.75 psig.

3.0 TECHNICAL EVALUATION

3.1 Adoption of NEI 94-01, Revision 2-A

As required by 10 CFR 50.54(o), the NMP2 primary reactor containment shall be subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the NMP2 10 CFR 50 Appendix J Testing Program Plan is based on RG 1.163, which endorses NEI 94-01, Revision 0, with certain modifications and additions. This license amendment request proposes to revise the NMP2 10 CFR 50 Appendix J Testing Program Plan by implementing the guidance in NEI 94-01, Revision 2-A. Revision 2-A of NEI 94-01 describes an approach for implementing the optional performance-based requirements of Option B, including provisions for extending Type A (ILRT) intervals to 15 years, and incorporates the regulatory positions stated in RG 1.163.

In the safety evaluation issued by NRC letter dated June 25, 2008 (Reference 3), the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and found that NEI 94-01, Revision 2, is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the safety evaluation. The following table addresses each of the six (6) limitations and conditions for NEI 94-01, Revision 2, listed in Section 4.1 of the NRC safety evaluation.

Limitation/Condition (from Section 4.1 of NRC safety evaluation)	NMP2 Response
1. For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SE Section 3.1.1.1).	Following NRC approval of this license amendment request, NMPNS will use the definition in Section 5.0 of NEI 94-01, Revision 2-A, for calculating the Type A leakage rate when future NMP2 Type A tests are performed.
2. The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3).	A schedule of containment inspections is provided in Section 3.2.2 of this Enclosure. Drywell interior coating inspections (Section 3.2.5.2 of this Enclosure) provide an additional opportunity to identify containment system structural problems.
3. The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3).	General visual examination of accessible interior and exterior surfaces of the containment system for structural problems is typically conducted in accordance with the NMP2 Containment Inservice Inspection (ISI) Plan and Schedule (Section 3.2.2 of this Enclosure), which implements the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g). The NMP2 containment system does not employ any moisture barriers and is not equipped with a sand cushion. There are no primary containment surface areas that require augmented examination in accordance with ASME Section XI, IWE-1240.
4. The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4).	There are no currently planned or anticipated major modifications to the NMP2 containment structure. The station design change process would address testing requirements for any future containment structure modifications.
5. The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2).	NMPNS acknowledges and accepts this NRC staff position, as communicated to the nuclear industry in Regulatory Issue Summary (RIS) 2008-27 dated December 8, 2008.

Limitation/Condition (from Section 4.1 of NRC safety evaluation)	NMP2 Response
6. For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable. NMP2 is not licensed under 10 CFR Part 52.

3.1.1 Evaluation of Technical Specification Changes

10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," states:

> "The regulatory guide or other implementation document used by a licensee or applicant for an operating license under this part or a combined license under part 52 of this chapter 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."

To comply with this requirement, TS 5.5.12.a currently references RG 1.163, which was issued in September 1995. RG 1.163 states that NEI 94-01, Revision 0, provides methods acceptable to the NRC for complying with Option B of 10 CFR 50, Appendix J, with four exceptions described therein.

The proposed change replaces the reference to RG 1.163 with a reference to NEI 94-01, Revision 2-A, dated October 2008. This proposed TS change is consistent with the regulatory requirement to include the implementation document used to develop the performance-based leakage testing program, by general reference, in the plant TS.

NMP2 TS 5.5.12.a currently lists two exceptions to the guidelines contained RG 1.163 (which references NEI 94-01, Revision 0). These exceptions were approved by the NRC in the safety evaluation that accompanied issuance of License Amendment No. 74 (Reference 11). The following discussion addresses these two exceptions.

<u>First Exception</u>

The first exception consists of two parts:

a. "The measured leakage of main steam isolation valves (MSIVs) is excluded from the combined leakage rate of 0.6 L_a."

   This represented an exception to NEI 94-01, Revision 0, Section 10.2, because treatment of MSIV leakage apart from L_a differed from the guidance in NEI 94-01, Revision 0, in that the combined leakage for Type B and C tests must meet the acceptance criterion of 0.6 L_a. The NRC accepted this exception on the basis that it was consistent with an existing exemption from the requirements of Appendix J authorized by NMP2 Operating License paragraph 2.D(ii)(b). The exemption was granted because MSIV leakage was treated separately from L_a in the radiological consequence analyses for the design basis LOCA.

The proposed change to TS 5.5.12.a retains this exception. NEI 94-01, Revision 2-A, Section 10.2, still indicates that the combined leakage for Type B and C tests must meet the acceptance criterion of 0.6 L_a (by reference to Section 6.4.4 of ANSI/ANS-56.8-2002 (Reference 21)), and the current radiological consequence analyses for the design basis LOCA that are based on the alternative source term (Reference 13) continue to treat MSIV leakage separately from L_a. The associated exemption from the requirements of Appendix J is included in NMP2 Renewed Facility Operating License paragraph 2.D(ii)(b). Thus, the basis for the previous NRC acceptance of this exception remains unchanged.

b. "As-found testing is not required to be performed on the MSIVs."

This also represented an exception to NEI 94-01, Revision 0, Section 10.2. The NRC accepted this exception on the basis that the MSIV test interval was not performance-based and the MSIV leakage was not included in L_a.

The proposed change to TS 5.5.12.a deletes this exception. Consistent with NEI 94-01, Revision 2-A, Section 10.2, the MSIV Type C test interval is non-performance based and is limited to 30 months. NMPNS intends to follow the guidance in Section 3.3.4 of ANSI/ANS-56.8-2002, which is referenced by NEI 94-01, Revision 2-A. This guidance indicates that an as-found test is required before work is done that can affect the leak rate of a component whose leakage integrity is suspect (i.e., has demonstrated poor reliability in maintaining an acceptably low leakage rate).

Second Exception

This exception states:

"Primary containment air lock door seals are tested prior to re-establishing primary containment OPERABILITY when something has been done that would bring into question the validity of the previous air lock door seal test."

This represented an exception to NEI 94-01, Revision 0, Section 10.2.2.1, which states only that air lock door seals must be tested prior to re-establishing containment integrity. This exception allows not testing the air lock door seals prior to re-establishing containment integrity when the reactor has been in a condition where containment integrity is not required but the air lock was not opened during the plant shutdown. The basis for this exception was that the air lock door seals will continue to perform their safety function if nothing has been done to invalidate the previous air lock door seal test.

The proposed change to TS 5.5.12.a retains this exception, since NEI 94-01, Revision 2-A, Section 10.2.2.1, still states only that air lock door seals must be tested prior to re-establishing containment integrity. The basis for the previous NRC acceptance of this exception remains unchanged.

3.1.2 Evaluation of Existing Exemptions from 10 CFR 50, Appendix J Requirements

10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," states:

> "Specific exemptions to Option A of this appendix that have been formally approved by the AEC or NRC, according to 10 CFR 50.12, are still applicable to Option B of this appendix, if necessary, unless specifically revoked by the NRC."

NMP2 implemented Option B of 10 CFR 50, Appendix J, via License Amendment No. 74 that was issued by NRC letter dated August 13, 1996 (Reference 11). At that time, evaluation of the need/basis for the exemptions listed in paragraph 2.D(ii) of the NMP2 Operating License was performed. The evaluation concluded that three of the four existing exemptions authorized by paragraph 2.D(ii) at that time continued to be necessary to support implementation of Option B. These were items (b) regarding MSIV leakage; (c) regarding the hydraulic control system for the reactor recirculation flow control valves; and (d) regarding the traversing incore probe system shear valves. The NRC safety evaluation that accompanied issuance of License Amendment No. 74 concluded that retaining these three prior exemptions was appropriate and consistent with the provisions of Option B of 10 CFR 50, Appendix J.

Paragraph 2.D(ii) of the NMP2 Renewed Facility Operating License, issued on October 31, 2006, continues to authorize the same three prior exemptions noted above. The proposed change to TS 5.5.12.a replaces the reference to RG 1.163 with a reference to NEI 94-01, Revision 2-A, as the implementation document used by NMPNS to develop the NMP2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. This change does not alter the need or basis for the three exemptions currently authorized by paragraph 2.D(ii) of the NMP2 Renewed Facility Operating License.

3.2 Extension of ILRT Interval to 15 Years

NEI 94-01, Revision 2-A, allows extensions of ILRT intervals based upon two consecutive, periodic successful Type A tests and the requirements stated in Section 9.2.3 of NEI 94-01. To support the proposed change to extend the ILRT interval to 15 years, the following evaluation presents NMP2 ILRT performance history, addresses each of the subsections in Section 9.2.3 of NEI 94-01, Revision 2-A, and discusses other considerations relating to maintaining containment integrity.

3.2.1 Type A Test Performance History

Acceptable performance history is defined as successful completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 L_a. A preoperational Type A test may be used as one of the two Type A tests that must be successfully completed to extend the test interval, provided that an engineering analysis is performed to document why a preoperational Type A test can be treated as a periodic test. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

As defined in NEI 94-01, Revision 2-A, the performance leakage rate is calculated as the sum of the Type A upper confidence limit (UCL) and as-left minimum pathway leakage rate (MNPLR) for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position (i.e., drained and vented to containment atmosphere) prior to performing the Type A test. In addition, leakage pathways that were isolated during performance of the test because of excessive leakage must be factored into the performance determination. The performance criterion for Type A tests is a performance leak rate of less than 1.0 L_a.

For the 1986 preoperational Type A test, the total time UCL leakage rate was 0.2815 wt% / day, excluding the minimum pathway leakage for isolated pathways. The minimum pathway leakage rate for Type B and C pathways not in service was 0.0087 wt% / day. There were no leakage pathways isolated during the performance of the test. Therefore, the performance leakage rate was 0.2815 + 0.0087 = 0.2902 wt% / day. The test was performed at peak accident pressure, P_a (39.75 psig).

For the 1991 periodic Type A test, the total time UCL leakage rate was 0.2880 wt% / day, excluding the minimum pathway leakage for isolated pathways. The minimum pathway leakage rate for Type B and C pathways not in service was 0.017 wt% / day. During the test, a leakage pathway through a containment pressure transmitter was isolated. Although no local leakage rate for this pathway was available, a maximum leakage through this pathway of 0.312 wt% / day was calculated. Therefore, the performance leakage rate was 0.2880 + 0.017 + 0.312 = 0.617 wt% / day. The test was performed at peak accident pressure, P_a (39.75 psig).

For the 2000 periodic Type A test, the total time UCL leakage rate was 0.2131 wt% / day, excluding the minimum pathway leakage for isolated pathways. The minimum pathway leakage rate for Type B and C pathways not in service was 0.0686 wt% / day. There were no leakage pathways isolated during the performance of the test. Therefore, the performance leakage rate was 0.2131 + 0.0686 = 0.2817 wt% / day. The test was performed at peak accident pressure, P_a (39.75 psig).

The above-described Type A test results were all less than the maximum allowable containment leakage rate (L_a at P_a) of 1.1% containment air weight per day at a pressure of 39.75 psig. This performance history supports extending the ILRT interval to 15 years.

3.2.2 Supplemental Inspection Requirements (NEI 94-01, Revision 2-A, Section 9.2.3.2)

Prior to initiating a Type A test, a general visual examination of accessible interior and exterior surfaces of the containment system for structural problems that may affect either the containment structure leakage integrity or the performance of the Type A test is performed. This inspection is typically conducted in accordance with the NMP2 Containment ISI Plan and Schedule (hereafter referred to as the IWE/IWL ISI program), which implements the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g). The applicable code edition and addenda for the second ten-year interval IWE/IWL ISI program is the 2001 Edition through 2003 Addenda of ASME Section XI. There are currently no relief requests associated with the second ten-year interval IWE/IWL ISI program. In the event that either a Subsection IWE or IWL examination is not scheduled to be performed during the same outage as the Type A test, a separate general visual inspection is required to be performed.

The examinations performed in accordance with the IWE/IWL ISI program satisfy the general visual examinations requirements specified in 10 CFR 50, Appendix J, Option B. Identification and evaluation of inaccessible areas are addressed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix)(A) and 10 CFR 50.55a(b)(2)(viii)(E). Examination of pressure-retaining bolted connections and evaluation of containment bolting flaws or degradation are performed in accordance with the requirements of 10 CFR 50.55a(b)(ix)(G) and 10 CFR 50.55a(b)(ix)(H). Each ten-year ISI interval is divided into three approximately equal-duration inspection periods. A minimum of one inspection during each inspection period of the ISI interval is required by the IWE/IWL ISI program. Since a 15-year ILRT interval spans at least four ISI inspection periods, the frequency of the examinations performed in accordance with the IWE/IWL ISI program satisfies the requirement of NEI 94-01, Revision 2-A, Section 9.2.3.2, to perform the general visual examinations during at least three other outages before the next Type A test if the Type

A test interval is to be extended to 15 years. This is illustrated by the following table showing the inspection periods for the NMP2 first and second ten-year IWE/IWL ISI intervals.

NMP2 Containment Inservice Inspection Periods (IWE/IWL)

Inspection Interval	Inspection Period	Period Start Date	Period End Date	Refuel Outage	Refuel Outage Year
1	1	April 5, 1998	April 4, 2001	RFO-07	2000
1	2	April 5, 2001	April 5, 2005	RFO-08 RFO-09	2002 2004
1	3	April 5, 2005	April 4, 2008	RFO-10 RFO-11	2006 2008
2	1	April 5, 2008	April 4, 2011	RFO-12	2010
2	2	April 5, 2011	April 4, 2015	RFO-13 RFO-14	2012 2014
2	3	April 5, 2015	April 4, 2018	RFO-15 RFO-16	2016 2018

The last Type A test was completed in April 2000 during refueling outage 07 (RFO-07). Based on a 15-year Type A test interval, the next Type A test would be scheduled for RFO-14 in 2014 (during Inspection Interval 2, Period 2). Thus, three containment system general visual examinations performed in accordance with the IWE/IWL ISI program take place prior to the 2014 Type A test (i.e., during Inspection Interval 1, Periods 2 and 3, and during Inspection Interval 2, Period 1).

There are no primary containment surface areas that require augmented examination in accordance with ASME Section XI, IWE-1240.

3.2.3 Deficiencies Identified During Supplemental Inspections (NEI 94-01, Revision 2-A, Section 9.2.3.3)

Consistent with the guidance provided in NEI 94-01, Revision 2-A, Section 9.2.3.3, abnormal degradation of the primary containment structure identified during the conduct of IWE/IWL ISI program examinations or at other times is entered into the corrective action program for evaluation to determine the cause of the degradation and to initiate appropriate corrective actions.

3.2.4 Plant-Specific Confirmatory Analyses (NEI 94-01, Revision 2-A, Section 9.2.3.4)

3.2.4.1 Methodology

An evaluation has been performed to assess the risk impact of extending the NMP2 containment ILRT interval from 10 years to 15 years. This plant-specific risk assessment followed the guidance in NEI 94-01, Revision 2-A (Reference 4), the methodology described in Electric Power Research Institute (EPRI) TR-1009325, Revision 2-A (Reference 10), and the NRC regulatory guidance outlined in RG 1.174 (Reference 9) on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change the licensing basis of the plant. In addition, the methodology used for Calvert Cliffs Nuclear Power Plant (Reference 14) to estimate the likelihood and risk implications of corrosion-induced leakage of steel containment liners going undetected during the extended ILRT interval was also used.

The current NMP2 Level 1 and Level 2 internal events PRA model was used to perform the plant-specific risk assessment. This PRA model has been updated to meet Capability Category II of ASME PRA Standard RA-Sb-2005 (Reference 15) and Regulatory Guide 1.200, Revision 1 (Reference 16). The model includes analyses for the dominant external events (seismic and fire), taken from the NMP2 Individual Plant Examination of External Events (IPEEE). Though the IPEEE seismic and fire event models have not been updated since the original IPEEE, they have been used to estimate the effect on total LERF of including these external events in the ILRT interval extension risk assessment.

In the safety evaluation issued by NRC letter dated June 25, 2008 (Reference 3), the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the safety evaluation. The following table addresses each of the four (4) limitations and conditions for the use of EPRI TR-1009325, Revision 2, listed in Section 4.2 of the NRC safety evaluation.

Limitation/Condition (from Section 4.2 of NRC safety evaluation)	**NMP2 Response**
1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension.	NMP2 PRA quality is addressed in Section 3.2.4.2 of this Enclosure.
2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this SE. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one-time ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point.	EPRI Report No. 1009325, Revision 2-A, incorporates these population dose and Conditional Containment Failure Probability (CCFP) acceptance guidelines, and these guidelines have been used for the NMP2 plant-specific risk assessment.
3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate for the pre-existing containment large leak rate accident case (accident case 3b) used by the licensees shall be 100 La instead of 35 La.	EPRI Report No. 1009325, Revision 2-A, incorporates the use of 100 L_a as the average leak rate for the pre-existing containment large leak rate accident case (accident case 3b), and this value has been used in the NMP2 plant-specific risk assessment.
4. A license amendment request (LAR) is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance.	NMP2 does not rely on containment overpressure to assure adequate net positive suction head for ECCS pumps following design basis accidents (see USAR Section 6.3.2.2).

3.2.4.2 PRA Quality

The NMP2 PRA is a Level 2 PRA that includes both internal and external events. Severe accident sequences have been developed from internally and externally initiated events, including internal floods, internal fires, and seismic events. The sequences have been developed to the radiological release end state; i.e., source term release to environment.

The NMP2 PRA is based on a detailed model of the plant developed from the Individual Plant Examination (IPE) and IPEEE projects, which underwent NRC review and the BWROG Certification Process. NRC reviews of the IPE and IPEEE are documented in the NRC staff evaluations dated August 18, 1994 for the IPE (Reference 17) and dated August 12, 1998 for the IPEEE (Reference 18). Review comments, current plant design, current procedures, plant operating data, current industry PRA techniques, and general improvements identified by the NRC have been incorporated into the current PRA model. The model is maintained in accordance with Constellation PRA procedures.

The NMP2 PRA internal events model has recently been updated to meet ASME PRA Standard RA-Sb-2005 (Reference 15) and RG 1.200, Revision 1 (Reference 16). The updated PRA model meets ASME Capability Category II requirements. The industry peer review of the updated PRA model has not yet been performed; however, a preliminary self-assessment of the model performed prior to the recent update identified gaps relative to RG 1.200 that were similar to those identified in the detailed self-assessment that was performed for the Nine Mile Point Unit 1 (NMP1) PRA model prior to its RG 1.200 update. The gaps identified for the NMP1 PRA were used to plan the NMP2 PRA model update. In addition, the NMP2 PRA update has been prepared by the same team of individuals that completed the recent NMP1 PRA update, which also meets ASME Capability Category II requirements. The NMP1 PRA update had few industry peer review team findings, and the peer review team commended NMPNS on the overall PRA model, supporting analyses, and documentation quality. A summary of the NMP1 PRA update peer review findings was submitted to the NRC by NMPNS letter dated December 4, 2008 (Reference 19). The findings were related primarily to specific documentation details and had an insignificant impact on the PRA results. These findings and other peer review suggestions have been considered in the NMP2 PRA update. As such, the updated NMP2 PRA model is considered acceptable for use in assessing the risk impact of extending the NMP2 containment ILRT interval from 10 years to 15 years.

3.2.4.3 Summary of Plant-Specific Risk Assessment Results

The findings of the NMP2 risk assessment confirm the general findings of previous studies (References 6 and 8) that the risk impact associated with extending the ILRT interval from 3 in 10 years to one in 15 years is small. The NMP2 plant-specific results are summarized below.

1. Core Damage Frequency (CDF) is not significantly impacted by the proposed change. NMP2 does not rely on containment overpressure to assure adequate net positive suction head for ECCS pumps following design basis accidents; thus, the CDF change is negligible and the relevant acceptance criterion is Large Early Release Frequency (LERF).

2. The increase in LERF based on consideration of internal events only is conservatively estimated as 7.3E-08/yr. The guidance in RG 1.174 defines very small changes in LERF as those that are less than 1E-07/yr. Therefore, the estimated change in LERF is determined to be very small using the guidelines of RG 1.174.

RG 1.174 also states that when the calculated increase in LERF is in the range of 1.0E-06 to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. An assessment of the impact from external events (seismic and fire) was also performed. In this case, the total increase in LERF for combined internal and external events was conservatively estimated as 1.01E-07, and the combined total LERF is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-05.

3. The calculated increase in the 50-mile population dose is 3.7E-02 person-rem per year. EPRI TR-1009325, Revision 2-A, states that a small increase in population dose is defined as an increase of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose (0.014 person-rem per year), whichever is less restrictive. Thus, the calculated 50-mile population dose increase is small using the guidelines of EPRI TR-1009325, Revision 2-A. Moreover, the risk impact when compared to other severe accident risks is negligible.

4. The calculated increase in the Conditional Containment Failure Probability (CCFP) is 0.87%. EPRI TR-1009325, Revision 2-A, states that increases in CCFP of less than or equal to 1.5 percentage points is very small. Therefore, the calculated CCFP increase is judged to be very small.

Details of the NMP2 risk assessment are contained in Attachment 3 to this Enclosure.

3.2.5 Additional Considerations

3.2.5.1 Type B and Type C Testing Program

The NMP2 Appendix J, Type B and Type C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the program as required by 10 CFR 50, Appendix J, Option B, and TS 5.5.12. The Type B and Type C testing program consists of local leak rate testing of penetrations with a resilient seal, expansion bellows, double-gasketed manways, hatches and flanges, drywell airlocks, and containment isolation valves that serve as a barrier to the release of the post-accident primary containment atmosphere. These components are tested with air or nitrogen at a pressure greater than or equal to 39.75 psig (P_a).

A review of the most recent Type B and Type C test results and their comparison with the allowable leakage rate specified in TS 5.5.12.d.1 was performed. The combined Type B and Type C leakage acceptance criterion (0.6 L_a) is 494.6 scfh. The maximum and minimum pathway leak rate summary totals for the last two refueling outages are shown below.

Refueling Outage	Maximum Pathway		Minimum Pathway	
	Leakage (scfh)	**% of 0.6 L_a (494.6 scfh)**	**Leakage (scfh)**	**% of 0.6 L_a (494.6 scfh)**
RFO11 - 2008	132.47	26.8%	82.3	16.6%
RFO10 - 2006	102.97	20.8%	85.9	17.4%

As discussed in NUREG-1493 (Reference 6), Type B and Type C tests can identify the vast majority (greater than 95%) of all potential primary containment leakage paths. This amendment request adopts the guidance in NEI 94-01, Revision 2-A, in place of NEI 94-01, Revision 0, but otherwise does not affect the scope, performance, or scheduling of Type B or Type C tests. Type B and Type C testing will continue to provide a high degree of assurance that primary containment integrity is maintained.

Frequently disassembled Type B penetrations (those with seals and gaskets, and bolted connections) are typically tested on a 30-month interval, whereas the test interval for infrequently disassembled Type B penetrations is typically 120 months (performance-based). Type C test intervals are performance-based (except for those valves on a fixed interval; e.g., MSIVs and feedwater isolation valves). Type C penetrations have had generally good performance and are typically tested on a 60-month interval. The Type B and Type C tests are scheduled such that approximately equal numbers of components are tested during each refueling outage, to levelize resource requirements.

3.2.5.2 Monitoring of Drywell Interior Coating

In addition to the inspections performed in accordance with the IWE/IWL ISI program, visual inspections of accessible interior surfaces of the drywell are performed each refueling outage to identify evidence of loose, flaking, or degraded painted surfaces. The suppression chamber is not included because it is primarily stainless steel and does not have Service Level 1 coatings. When degraded coatings are identified, evaluations are performed to determine any necessary actions (e.g., repair, removal, or replacement). These inspections provide another opportunity to identify containment system structural problems.

3.2.5.3 NRC Information Notice 92-20, Inadequate Local Leak Rate Testing

NRC Information Notice 92-20 was issued to alert licensees to problems with local leak rate testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem.

The utilization of bellows as containment pressure retaining boundaries is limited to the following penetrations:

- 2NMT*Z31A, B, C, D and E: Traversing Incore Probe (TIP) drive guide tubes to reactor vessel.

Type B tests are applicable to these penetration bellows. A makeup pressure test is utilized to determine primary containment penetration leak rates. In this test, the TIP tubing is disconnected and, using a test fixture, penetrations 2NMT*Z31A, B, C, D and E are tested together with their associated TIP solenoid-operated ball valve at a test interval of 24 months (based on inservice testing program requirements).

The NMP2 plant-specific risk assessment provided as Attachment 3 to this Enclosure takes into consideration the potential failure of containment bellows assemblies.

3.2.5.4 Aging Management Examination of Containment Penetration Bellows

An augmented VT-1 visual examination of the containment penetration bellows will be performed using enhanced techniques qualified for detecting stress corrosion cracking, per NUREG-1611. This is an addition to the IWE/IWL ISI program and has been reviewed and accepted by the NRC as part of the License Renewal application review (Reference 20). These inspections are beyond the scope of examinations required by ASME Section XI, Table IWE-2500-1, and are not considered augmented examinations as defined in ASME Section XI, IWE-1240.

3.2.5.5 Plant Operational Performance

During power operation, the NMP2 primary containment is inerted with nitrogen to maintain oxygen concentration within TS 3.6.3.2 limits. As a result, the primary containment is maintained at a slightly positive pressure. Drywell pressure is continuously recorded and is verified to be within limits by TS Surveillance Requirement 3.6.1.4.1 every 12 hours. Maintaining the containment pressurized at power and frequently monitoring drywell pressure assures that gross containment leakage that may develop during power operation will be detected.

3.3 Conclusions

NEI 94-01, Revision 2-A, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163, and includes provisions for extending Type A (ILRT) intervals to 15 years. NEI 94-01, Revision 2-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. NMPNS is adopting the guidance of NEI 94-01, Revision 2-A, for the NMP2 10 CFR 50 Appendix J Testing Program Plan described in TS 5.5.12, with two exceptions that were previously reviewed and accepted by the NRC in License Amendment No. 74. Existing exemptions from the requirements of Appendix J authorized by paragraph 2.D(ii) of the NMP2 Renewed Facility Operating License continue to be necessary to support implementation of Option B of Appendix J.

Based on the previous ILRT tests conducted at NMP2, which confirm that the primary containment structure exhibits extremely low leakage, NMPNS concludes that extension of the containment ILRT interval from 10 to 15 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Option B of 10 CFR 50 Appendix J, inspection activities performed as part of the plant IWE/IWL ISI program, inspections of drywell interior coatings, and by operating experience with a containment that normally operates at a positive pressure (i.e., the pressure from containment inerting). In the aggregate, these provide continuing confidence in containment integrity.

This experience is supplemented by risk analysis studies, including the NMP2 risk analysis provided in Attachment 3 to this Enclosure. The findings of the NMP2 risk assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from 10 to 15 years results in a very small change to the NMP2 risk profile.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.54(o) states that primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J of 10 CFR 50.

10 CFR Part 50, Appendix J, Option B, requires that licensees' primary reactor containments meet the leakage rate requirements as delineated by Appendix J. This requirement is met by performance of Type A, B, and C leakage rate testing on the primary containment and its associated components (e.g., valves, penetrations). The leakage rate test results are compared to allowable leakage rate acceptance criteria set forth in Appendix J.

NEI 94-01, Revision 2-A, describes an approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163, and includes provisions for extending Type A (ILRT) intervals to 15 years. NEI 94-01, Revision 2-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. In the safety evaluation issued by NRC letter dated June 25, 2008 (Reference 3), the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the safety evaluation.

EPRI TR-1009325, Revision 2, provides a risk impact assessment for optimized ILRT intervals up to 15 years, utilizing current industry performance data and risk-informed guidance. NEI 94-01, Revision 2-A, states that a plant-specific risk impact assessment should be performed using the approach and methodology described in TR-1009325, Revision 2, for a proposed extension of the ILRT interval to 15 years. In the safety evaluation issued by NRC letter dated June 25, 2008, the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, is acceptable for referencing by licensees proposing to amend their TS to extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the safety evaluation.

The NMP2 10 CFR 50 Appendix J Program Plan will continue to comply with the requirements of 10 CFR 50, Appendix J. The proposed amendment is consistent with the NRC-accepted guidance in NEI 94-01, Revision 2-A, and ERPI TR-1009325, Revision 2-A.

4.2 Significant Hazards Consideration

Nine Mile Point Nuclear Station, LLC (NMPNS) is requesting revisions to Nine Mile Point Unit 2 (NMP2) Technical Specification (TS) 5.5.12, "10 CFR 50 Appendix J Testing Program Plan." The proposed amendment would replace the reference to Regulatory Guide (RG) 1.163 with a reference to Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 2-A, dated October 2008, as the implementation document used by NMPNS to develop the NMP2 performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. The proposed amendment would also extend the interval for the primary containment integrated leak rate test (ILRT), which is required to be performed by 10 CFR 50, Appendix J, from 10 years to no longer than 15 years from the last ILRT.

NMPNS has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No.

   The proposed amendment involves changes to the NMP2 10 CFR 50 Appendix J Testing Program Plan. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for development of the NMP2 performance-based leakage testing program. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less the values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval from 10 years to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small, and the increase in the large early release frequency resulting from the proposed change was determined to be within the guidelines published in NRC RG 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. NMPNS has determined that the increase in conditional containment failure probability due to the proposed change would be very small. Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for development of the NMP2 performance-based leakage testing program, and establishes a 15 year interval for the performance of the primary containment ILRT. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

   Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for development of the NMP2 performance-based leakage testing program, and establishes a 15 year interval for the performance of the primary containment ILRT. This amendment does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the 10 CFR 50 Appendix J Testing Program Plan, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant safety analyses is maintained.

The overall containment leakage rate limit specified by the TS is maintained, and the Type A, B, and C containment leakage tests will continue to be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 2-A.

Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by an ILRT. In addition, the on-line containment monitoring capability that is inherent to inerted boiling water reactor containments allows for the detection of gross containment leakage that may develop during power operation. This combination of factors ensures that evidence of containment structural degradation is identified in a timely manner. Furthermore, a risk assessment using the current NMP2 Probabilistic Risk Assessment model concluded that extending the ILRT test interval from 10 years to 15 years results in a very small change to the NMP2 risk profile.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, NMPNS concludes that the proposed change presents no significant hazards considerations under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change surveillance requirements regarding leak rate testing of the primary containment. However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

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

2. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 2, August 2007

3. Letter from M. J. Maxin (NRC) to J. C. Butler (NEI) dated June 25, 2008, Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (TAC No. MC9663)

4. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 2-A, October 2008

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

6. NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995

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

8. EPRI TR-1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 2, 2007

9. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002

10. EPRI TR-1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 2-A, October 2008

11. Letter from D. S. Hood (NRC) to B. R. Sylvia (NMPC) dated August 13, 1996, "Issuance of Amendment for Nine Mile Point Nuclear Station, Unit 2 (TAC No. M94641)"

12. Letter from G. S. Vissing (NRC) to J. H. Mueller (NMPC) dated February 15, 2000, "Conversion to Improved Technical Specifications for Nine Mile Point Nuclear Station Unit No. 2 – Amendment No. 91 to Facility Operating License No. NPF-69 (TAC No. MA3822)"

13. Letter from R. V. Guzman (NRC) to K. J. Polson (NMPNS) dated May 29, 2008, "Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment Re: Implementation of Alternative Radiological Source Term (TAC No. MD5758)"

14. Letter from C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to Document Control Desk (NRC) dated March 27, 2002, "Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension"

15. ASME RA-Sb-2005, "Addenda to ASME RA-S-2002, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," December 30, 2005

16. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Assessment Results for Risk-Informed Activities," Revision 1, January 2007

17. Letter from P. T. Kuo (NRC) to B. R. Sylvia (NMPC) dated August 18, 1994, "Individual Plant Examination for Nine Mile Point Nuclear Station, Unit 2 (TAC No. M74437)"

18. Letter from D. S. Hood (NRC) to J. H. Mueller (NMPC) dated August 12, 1998, "Review of Individual Plant Examination of External Events, Nine Mile Point Nuclear Station, Unit No. 2 (TAC No. M83646)"

19. Letter from K. J. Polson (NMPNS) to Document Control Desk (NRC) dated December 4, 2008, "License Amendment Request Pursuant to 10 CFR 50.90: One-Time Extension of the Primary Containment Integrated Leakage Rate Test Interval – Response to NRC Request for Additional Information (TAC No. MD9453)"

20. NUREG-1900, "Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2," September 2006

21. ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements," November 27, 2002

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGE (MARK-UP)

The current version of Technical Specification Page 5.5-11 has been marked-up by hand to reflect the proposed change.

5.5 Programs and Manuals

5.5.12 10 CFR 50 Appendix J Testing Program Plan (continued)

Insert A

Section 2.D(ii) of the Operating License. ~~This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled, "Performance-Based Containment Leak Test Program," dated September 1995 with the following exceptions:~~

1. The measured leakage of main steam isolation valves (MSIVs) is excluded from the combined leakage rate of 0.6 L_a~~, and as-found testing is not required to be performed on the MSIVs~~.

2. Primary containment air lock door seals are tested prior to re-establishing primary containment OPERABILITY when something has been done that would bring into question the validity of the previous air lock door seal test.

b. The peak calculated containment internal pressure (P_a) for the design basis loss of coolant accident is 39.75 psig.

c. The maximum allowable primary containment leakage rate (L_a) at P_a shall be 1.1% of primary containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Primary Containment leakage rate acceptance criterion is < 1.0 L_a. The combined leakage rate for Type B and C tests on a minimum pathway basis, except for main steam line isolation valves and Primary Containment isolation valves which are hydrostatically tested, is $\leq$ 0.6 L_a.

   During the first unit startup following testing in accordance with this program, the as-left combined leakage rate acceptance criteria are < 0.6 L_a for the Type B and C tests on a maximum pathway basis, except for main steam line isolation valves and Primary Containment isolation valves which are hydrostatically tested, and $\leq$ 0.75 L_a for Type A tests.

2. Air lock testing acceptance criteria are:

   (a) Overall air lock leakage rate is $\leq$ 0.05 L_a when tested at greater than or equal to P_a; and

(continued)

INSERT A (for TS Page 5.5-11)

This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008, with the following exceptions:

ATTACHMENT 2

RISK IMPACT ASSESSMENT OF EXTENDING CONTAINMENT TYPE "A" TEST INTERVAL FOR NMP2

(2NER-PR-003)

Nine Mile Point Nuclear Station, LLC
August 13, 2009

Constellation Energy Group

Nine Mile Point Nuclear Station

NUCLEAR ENGINEERING REPORT
NINE MILE POINT UNIT 2

Risk Impact Assessment of Extending Containment Type "A" Test Interval for NMP2

2NER-PR-003

Prepared By:	J H Moody	Date: 6/24/09
Reviewed By:	Hiep Huynh	Date: 6/24/09
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List of Tables and Figures

1 Purpose of the Analysis

1.1 Purpose

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A (integrated leak rate test - ILRT) interval to a permanent fifteen years. The extension would allow for substantial cost savings as the Type A test could be deferred for additional scheduled refueling outages for Nine Mile Point Unit 2 (NMP2). The risk assessment follows the guidelines from NEI 94-01, Revision 2-A [1], the methodology described in EPRI 1009325 Revision 2-A [27], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to Type A test interval extensions, and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4]. In addition, the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [5] has been used as incorporated in the EPRI methodology.

1.2 Background

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Type A surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A test frequency is 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 rate was less than the limiting containment leakage rate of 1La. The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995 [6], provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."[2]

NUREG-1493 [6] analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative BWR plant (i.e., Peach Bottom), increasing the containment leakage rate from the nominal 0.5 percent per day to 5 percent per day results in a small increase in total population exposure. In addition, increasing the leakage rate to 50 percent per day increases the total population risk by less than 1 percent. Consequently, it is desirable to show that extending the Type A test interval will not lead to a significant increase in risk for NMP2.

The Guidance provided in Appendix H of EPRI 1009325 Revision 2-A [27] (also identified as EPRI 1018243) for performing risk impact assessments in support of Type A test extensions builds on the EPRI Risk Assessment methodology described in EPRI TR-104285. This methodology of EPRI 1009325 is followed to determine the appropriate risk information for use in evaluating the impact of the proposed Type A test interval changes.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. 10CFR50.55a(b)(2)(ix)(E) requires that a general visual examination as required by Subsection IWE must be performed once each inspection interval. The guidance in NEI 94-01 Revision 2-A indicates that general visual examinations must be conducted prior to each Type A test and at least three other outages before the next Type A test if the test interval is being extended to 15 years. These requirements will not be changed as a result of the extended Type A test interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets, and Type C local leak test performed to verify leak-tight integrity if containment isolation valves are also not affected by the change to the Type A test frequency.

1.3 Criteria

The acceptance guidelines in RG 1.174 [4] are used to assess the acceptability of this extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 1E-6 per reactor year and increases in large early release frequency (LERF) less than 1E-7 per reactor year. Since the Type A test interval does not significantly impact CDF, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 1E-6 per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability (CCFP) that helps to ensure that the defense-in-depth philosophy is maintained is also calculated.

Regarding CCFP, changes of up to 1.1% have been accepted by the NRC for the one-time requests for extension of Type A test intervals, and a CCFP of 1/10 (10%) has been approved for application to evolutionary light water designs [27]. Based on the criteria stated in EPRI 1009325 Revision 2-A, a change in the CCFP of up to 1.5% is considered to be small.

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter. Based on the criteria stated in EPRI 1009325 Revision 2-A, a very small population dose is defined as an increase from the baseline Type A test interval (3 tests per 10 years) dose of $\leq$ 1.0 person-rem per year or 1% of the total baseline dose, whichever is less restrictive for the risk impact assessment of the proposed extended Type A test interval.

In addition, EPRI 1009325 Revision 2-A requires that ECCS NPSH requirements be assessed with regard to determining whether containment over pressure is required in various accident scenarios. The NMP2 ECCS pumps are capable of pumping saturated fluids and containment over pressure is not required in the PRA for ECCS pumping success. As a result, Type A test interval changes have an insignificant impact on CDF and LERF is the proper risk metric to be considered.

2 Methodology

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the NMP2-specific change in risk associated with increasing the Type A test interval to fifteen years. EPRI 1009325 Revision 2-A [27] provides a generally applicable assessment of the risk involved in extension of Type A test intervals to permanent 15-year intervals. Appendix H of the EPRI report provides guidance for performing plant-specific supplemental risk impact assessments. The approach included in this guidance document is used in the NMP2 assessment to determine the estimated increase in risk associated with the Type A test interval extension. This EPRI document includes the bases for the values assigned in determining the probability of leakage for the EPRI Class 3a and 3b scenarios in this analysis as described in Section 5.

This analysis uses results from a Level 2 analysis of core damage scenarios from the current NMP2 PRA model and subsequent containment response resulting in various fission product release categories (including no or negligible release).

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.

2. Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.

3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the Type A test interval to fifteen years.

4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [4] and compare with the acceptance guidelines of RG 1.174.

5. Determine the impact on the Conditional Containment Failure Probability (CCFP).

6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, external events, and the fractional contribution to LERF of increased large isolation failures (due to liner breach).

3 Ground Rules

The following ground rules are used in the analysis consistent with EPRI 1009325 Revision 2-A:

- The technical adequacy of the NMP2 PRA is consistent with the requirements of Regulatory Guide 1.200 Rev 1 as is relevant to this Type A test interval extension.
- The NMP2 Level 1 and Level 2 internal events PRA models provide representative results.
- It is appropriate to use the NMP2 internal events PRA model as a gauge to effectively describe the risk change attributable to the Type A test interval extension. It is reasonable to assume that the impact from the extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations.
- Dose results for the containment failures modeled in the PRA can be characterized by information provided in NUREG/CR-4551 [7]. They are estimated by scaling the NUREG/CR-4551 results by population differences for Nine Mile Point Station compared to the NUREG/CR-4551 reference plant. Using this reference plant is judged reasonable as it was also used for another Mark II BWR as an example in EPRI 1009325 Revision 2-A.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is 10La.
- The representative containment leakage for Class 3b sequences is 100La.
- The Class 3b sequences can be conservatively categorized as LERF.
- The impact on population doses from containment bypass scenarios is not altered by the proposed Type A test interval extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this separate categorization.
- The reduction in Type A test frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The NMP2 PRA does contain external events included from the IPEEE; however, these event models have not been updated and maintained. Still, the sensitivity of the Class 3b

contribution to LERF from external events is evaluated using the NMP2 PRA to provide an order of magnitude estimate for contribution of external events to the impact of the changed Type A test interval.

4 Inputs

This section summarizes general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 General Resources

Various industry studies on containment leakage risk assessment are available and support the final methodology in EPRI 1009325 Revision 2-A [27]. Each of the following documents is summarized in further detail in EPRI 1009325:

1. NUREG/CR-3539 [10]
2. NUREG/CR-4220 [11]
3. NUREG-1273 [12]
4. NUREG/CR-4330 [13]
5. EPRI TR-105189 [14]
6. NUREG-1493 [6]
7. EPRI TR-104285 [2]
8. NUREG-1150 [15] and NUREG/CR-4551 [7]
9. NEI Interim Guidance [3][20]
10. Calvert Cliffs liner corrosion analysis [5]
11. EPRI Report No. 1009325, Revision 2-A, Appendix H [27]

4.2 Plant Specific Inputs

The plant-specific information used to perform the NMP2 Type A test interval extension risk assessment includes the following:

- Level 1 Model results [17]
- Level 2 Model results [17]
- Release category definitions used in the Level 2 Model [18]
- Population within a 50-mile radius [19]
- Type A test results demonstrate adequacy of the administrative and hardware issues [28]. The two most recent Type A tests at NMP2 have been successful, so the current Type A test interval requirement is 10 years.
- Containment failure probability data [18]

<u>Level 1 Model</u>
The Level 1 PRA model that is used for NMP2 is characteristic of the as-built plant. The current Level 1 model is a linked fault tree model, and was quantified with a total Core Damage Frequency (CDF) = 8.4E-6/yr (1E-12 truncation).

Level 2 Model

The Level 2 Model that is used for NMP2 was developed to calculate the LERF contribution (Release Category H-E) as well as the other release categories evaluated in the model. Table 1 summarizes the NMP2 Level 2 results (1E-13 truncation) in terms of release category [17, 18].

Table 1: Level 2 PRA Release Categories and Frequencies

Release Category	Definition of Release	Events/yr
OK	Containment Intact	3.09E-06
LL-L	Low-Low Magnitude and Late Timing	2.49E-08
LL-I	Low-Low Magnitude and Intermediate Timing	1.74E-06
LL-E	Low Magnitude and Early Timing	9.55E-10
L-L	Low Magnitude and Late Timing	0.00
L-I	Low Magnitude and Intermediate Timing	2.50E-07
L-E	Low Magnitude and Early Timing	1.51E-06
M-L	Medium Magnitude and Late Timing	0.00
M-I	Medium Magnitude and Intermediate Timing	8.23E-07
M-E	Medium Magnitude and Early Timing	2.48E-07
H-L	High Magnitude and Late Timing	0.00
H-I	High Magnitude and Intermediate Timing	2.34E-07
H-E	High Magnitude and Early Timing	4.43E-07
	Total Release Category Frequency	**8.37E-06**

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for NMP2. Each of the release categories from Table 1 is associated with an applicable Collapsed Accident Progression Bin (APB) from NUREG/CR-4551 (see below). The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 10 bins that are relevant to the analysis. The definitions of the 10 collapsed APBs are provided in NUREG/CR-4551 [7] and are reproduced in Table 2 for references purposes. Table 3 summarizes the calculated population dose for Peach Bottom associated with each APB from NUREG/CR-4551 [7]. The EPRI approach was used in lieu of the level 3 PRA model that was developed for the NMP License Renewal Application [29] because the level 3 model has not been maintained since its creation. It has not been updated to the same level of quality as the level 1 and level 2 PRA (RG 1.200) and has not been updated since the License Renewal Application for which it was developed.

Table 2: Summary Accident Progression Bin (APB) Descriptions

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

Table 3: Calculation of Peach Bottom Population Dose Risk at 50 Miles

Collapsed Accident Progression (APB) Number	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (From a total of 7.9 person-rem/yr, mean) (2)	NUREG/CR-4551 Collapsed Bin Frequencies (per year) (3)	NUREG/CR-4551 Population Dose at 50 miles (Person-rem) (4)
1	0.021	0.1659	9.55E-8	1.74E+6
2	0.0066	0.05214	4.77E-8	1.09E+6
3	0.556	4.3924	1.48E-6	2.97E+6
4	0.226	1.7854	7.94E-7	2.25E+6
5	0.0022	0.01738	1.30E-8	1.34E+6
6	0.059	0.4661	2.04E-7	2.28E+6
7	0.118	0.9322	4.77E-7	1.95E+6
8	0.0005	0.00395	7.99E-7	4.94E+3
9	0.01	0.079	3.86E-7	2.05E+5
10	0	0	4.34E-8	0
Totals	1.0	7.9	4.34E-6	

(1) Mean Fractional Contribution to Risk calculated from Table 5.2-3 of NUREG/CR-4551

(2) The total population dose risk (PDR) at 50 miles from internal events in person-rem is provided in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.

(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-6. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.

(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.

The person-rem results in Table 3 can be used as an approximation of the dose for the NMP2 site if it is corrected for allowable containment leak rate (La), reactor power level and the population density surrounding NMP2 [27].

- Leak rate adjustment = La of NMP2 (%w/o/day) ÷ La of Peach Bottom
  = 1.1 ÷ 0.5
  = 2.20
  La for Peach Bottom is 0.5%w/o/day
  La for NMP2 is 1.1%w/o/day based on Technical Specification 5.5.12
  This is applicable only to those APBs affected by normal leakage

- Power level adjustment = Proposed upated power at NMP2 (MWt) ÷ Rated power at Peach Bottom
  = 3988 MWt ÷ 3293 MWt
  = 1.211
  The rated power level for Peach Bottom is 3293 MWt
  The proposed uprated power level for NMP2 is 3988 MWt

- Population density adjustment = NMP2 population ÷ Peach Bottom population
  = 914,688 / 3.2E+06
  = 0.286

The total population within a 50-mile radius of NMP2 is 914,668 [19]. This population value is compared to the population value that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for NMP2. Peach Bottom Population from NUREG/CR-4551 is 3.2E+06, as referenced in the EPRI 1009325 Rev 2-A.

The factors developed above are used to adjust the population dose for the surrogate plant (Peach Bottom) for NMP2. For intact containment end states, the total population dose factor is as follows:

FIntact = FPopulation * FPower Level * FLeakage

FIntact = 0.286 * 1.211 * 2.20

FIntact = 0.76

For EPRI accident classes not dependent on containment leakage, the population dose factor is as follows:

FOthers = FPopulation * FPower Level

FOthers = 0.286 * 1.211

FOthers = 0.35

The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. The above adjustments provide an approximation for NMP2 of the population doses associated with each of the release categories from NUREG/CR-4551.

Table 4 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for NMP2.

Table 4: Calculation of NMP2 Population Dose Risk at 50 Miles

Peach Bottom Accident Progression Bin #	NUREG/CR-4551 Population Dose at 50 miles (Person-rem)	Bin Multiplier used to obtain NMP2 Population Dose	NMP2 Adjusted Population Dose at 50 miles (Person-rem)
1	1.74E+6	0.35	6.09E+5
2	1.09E+6	0.35	3.82E+5
3	2.97E+6	0.35	1.04E+6
4	2.25E+6	0.35	7.88E+5
5	1.34E+6	0.35	4.69E+5
6	2.28E+6	0.35	7.98E+5
7	1.95E+6	0.35	6.83E+5
8	4.94E+3	0.76	3.75E+3
9	2.05E+5	0.35	7.18E+4
10	0	0.35	0.0

Application of NMP2 PRA Results to NUREG/CR-4551 Level 3 Output

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the NMP2 PRA Level 2 model are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in that document, it was necessary to match the NMP2 PRA Level 2 release categories to the collapsed APBs. The assignments are shown in Table 5, along with the corresponding EPRI classes (see below).

Table 5: EPRI Class Dose and Frequency Assignment

EPRI Class	EPRI Description	Leakage Basis	Peach Bottom APB for Dose (NUREG/CR-4551)	NMP2 Level 2 Release Category Frequency (or other)
1	No Containment Failure	La	8	OK (Intact) – 3a – 3b
2	Large Containment Isolation Failure	Plant value	3 (highest dose)	Containment Isolation failure (H-E with IS=F)
3a	Small pre-existing failure	10 La	10 x Dose of APB 8	By methodology (CDF-(H-E)*0.0092)
3b	Large pre-existing failure	100 La	100 x Dose of APB 8	By methodology (CDF-(H-E)*0.0023)
4	Small Isol failure – Type B	NA	NA	NA
5	Small Isol failure – Type C	NA	NA	NA
6	Cont Isol failure – Dep failure	NA	NA	NA
7	Severe Accident Sequences	Plant value (weighted average)	Weighted Average	Weighted Average of Subcategories (7a, 7b, 7c, 7d, 7e)
7a	Subcategory (not EPRI)		3 (highest dose)	H-E without IS=F and V
7b	Subcategory (not EPRI)		6 (high dose, late DW failure)	H-M & H-L
7c	Subcategory (not EPRI)		1 (WW failure early)	M-E
7d	Subcategory (not EPRI)		2 (WW failure late)	M-I & M-L
7e	Subcategory (not EPRI)		9 (CD, no Vessel Breach)	L & LL
8	Containment Bypass	Plant value	3 (highest dose)	H-E Class V Scenarios

Release Category Definitions

Table 6 defines the accident classes used in the Type A test interval extension evaluation, which are consistent with the EPRI methodology [27]. These containment failure classifications are used in this analysis to determine the risk impact of extending the containment Type A test interval as described in Section 5 of this report.

Table 6: EPRI Containment Failure Classification

Class	Description
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
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.

Class	Description
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.
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.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
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 impact these accidents.

4.3 Impact of Extension on Detection of Failures that Lead to Leakage

The Type A test can detect a number of component failures such as liner breach, failure of certain bellows arrangements and failure of some sealing surfaces, which can lead to leakage. The proposed Type A test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class as defined in Table 6 is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the EPRI Guidance [27]. For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., 2 "small" failures in 217 tests leads to 2/217 = 0.0092). For Class 3b, Jeffreys non-informative prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5 / (217+1) = 0.0023).

In a follow on letter [20] to their interim guidance document [3], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of NRC Regulatory Guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> *The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage.*

The application of this additional guidance to the analysis for NMP2, as detailed in Section 5, involves the following:

- The LERF sequences (Class 2, Class 8 and H-E portion of Class 7) are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the NMP2 Level 2 PRA analysis.

- Class 1 accident sequences may involve availability and or successful operation of containment sprays. It could be assumed that, for calculation of the Class 3b and 3a frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. However, this has been conservatively neglected in this evaluation. Also, other sequences that have the potential to never be LERF (e.g., late core damage) have conservatively not been deleted from Class 3b.

Consistent with the methodology [27], the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yr / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yr / 2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by Type A testing. Correspondingly, an extension of the Type A test interval to fifteen years can be estimated to lead to a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is conservative compared to previous submittals (e.g., the IP3 request for a one-time Type A test interval extension that was approved by the NRC [9]) because it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur). Eliminating this possibility conservatively over-estimates the factor increases attributable to the Type A test interval extension.

4.4 Impact of Extension on Detection of Steel Liner Corrosion

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and going undetected during the extended Type A test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. The NMP2 containment is a pressure-suppression BWR Mark II type. The drywell is a steel-lined reinforced concrete vessel, and the suppression chamber is a stainless steel clad, steel-lined reinforced concrete vessel.

The following approach is used to determine the change in likelihood, due to extending the Type A test interval, of detecting corrosion of the containment steel liner. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

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

Assumptions

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures (see Table 7, Step 1).
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to the NMP2 containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2 were initiated from the non-visible (backside) portion of the containment liner (see Table 7, Step 1).
- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also 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 have been performed since the time frame of the Calvert Cliffs analysis), and there is no evidence that additional corrosion issues were identified (See Table 7, Step 1).
- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages (See Table 7, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every ten years and every two years.
- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the probability versus containment pressure, and the selected values are consistent with a pressure that corresponds to the Type A test target pressure of ~40 psig. For NMP2, the containment failure probabilities are less than these values at 40 psig [18]. Sensitivity studies are included that increase and decrease the probabilities by an order of magnitude (See Table 7, Step 4).

- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than the containment cylinder and dome region (See Table 7, Step 4).

- Consistent with the Calvert Cliffs analysis, 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 likelihood of 5% and 15%, respectively.

- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Table 7: Steel Liner Corrosion Base Case

<table>
<tr><th>Step</th><th>Description</th><th colspan="2">Containment Cylinder and Dome</th><th colspan="2">Containment Basemat</th></tr>
<tr><td>1</td><td>Historical Steel Liner Flaw Likelihood
Failure Data: Containment location specific (consistent with Calvert Cliffs analysis)</td><td colspan="2">Events: 2
2/(70 * 5.5) = 5.2E-3</td><td colspan="2">Events: 0 (assume half failure)
0.5/(70 * 5.5) = 1.3E-3</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Age Adjusted Steel Liner Flaw Likelihood
During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis)</td><td>Year
1
avg 5-10
15</td><td>Failure Rate
2.1E-3
5.2E-3
1.4E-2</td><td>Year
1
avg 5-10
15</td><td>Failure Rate
5.1E-4
1.3E-3
3.6E-3</td></tr>
<tr><td colspan="2">15 year average = 6.4E-3</td><td colspan="2">15 year average = 1.6E-3</td></tr>
<tr><td>3</td><td>Flaw Likelihood at 3, 10, and 15 years
Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference [5]).</td><td colspan="2">0.71% (1 to 3 years)
4.14% (1 to 10 years)
9.66% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the desired presentation of the results.</td><td colspan="2">0.18% (1 to 3 years)
1.03% (1 to 10 years)
2.41% (1 to 15 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with desired presentation of the results.</td></tr>
</table>

Step	Description	Containment Cylinder and Dome	Containment Basemat
4	Likelihood of Breach in Containment Given Steel Liner Flaw The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1%, (compared to 0.11% in the Calvert Cliffs analysis).	1%	0.1%
5	Visual Inspection Detection Failure Likelihood Utilizes assumptions consistent with Calvert Cliffs analysis	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by Type A test) 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)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0097% (at 15 years)** 9.7% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 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 as summarized below for NMP2.

Total Likelihood of Non-Detected Containment Leakage Due To Corrosion for NMP2:
At 3 years: 0.00071% + 0.00018% = 0.00089%
At 10 years: 0.0041% + 0.0010% = 0.0052%
At 15 years: 0.0097% + 0.0024% = 0.012%

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF. For example, the 3-in-10 year case is calculated as follows:

- Per Table 9, the EPRI Class 3b frequency is 1.82E-8/yr. As discussed in Section 5.1, this is the NMP2 CDF associated with accidents that are not independently LERF [CDF – (H-E) = 7.93E-6] times the conditional probability of Class 3b (0.0023).

- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 7.93E-6 [CDF – (H-E)] * 8.9E-6 = 7.1E-11/yr, where 8.9E-6 as shown above is the cumulative likelihood of non-detected containment leakage due to corrosion at 3 years.

- The 3-in-10 year Class 3b frequency including the corrosion-induced concealed flaw issue is calculated as 1.82E-8/yr + 7.1E-11/yr= 1.83E-8/yr.

5 Results

The application of the approach based on the guidance contained in Appendix H of EPRI 1009325 Revision 2-A [27] and previous risk assessment submittals on this subject [5, 8, 21, 22, 23] have led to the following results described in this section. The results are displayed according to the eight accident classes defined in the EPRI report. Table 8 lists these accident classes.

The analysis performed examined NMP2-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the break down of the severe accidents contributing to risk was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI 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 bellows leakage (EPRI Class 3 sequences).

- 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 Class 6 sequences). This class is not specifically examined since it will not significantly influence the results of this analysis.

- Accident sequences involving containment bypassed (EPRI Class 8 sequences), large containment isolation failures (EPRI Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the Type A test interval change.

- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 8: EPRI Accident Classes

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal –Type B)
5	Small Isolation Failures (Failure to seal—Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including very low and no release)

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

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 8.

Step 2 - Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

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

Step 5 - Determine the impact on the Conditional Containment Failure Probability (CCFP)

5.1 Step 1 – Quantify the Base-Line Risk

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.

For the assessment of Type A test interval impacts on the risk profile, the potential for pre-existing leaks is included in the model (These events are represented by the EPRI Class 3 sequences). The question on containment integrity is modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes are considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies for the severe accident classes defined in Table 8 are developed for NMP2 by first determining the frequencies for Classes 1, 2, 7 and 8 using the categorized sequences and the identified correlations shown in Table 5, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4.

The total frequency of the categorized sequences is 8.4E-6/yr the same as total CDF. Table 9 contains the frequencies from the categorized sequences. The results are summarized below and in Table 10.

Table 9: NMP2 Categorized Accident Classes and Frequencies

EPRI Class	NMP2 Frequency (per yr)	NMP2 Basis (release category)
1	3.00E-06	OK – EPRI 3a – EPRI 3b
2	2.95E-07	IS=F contribution from H-E
3a	7.29E-08	[CDF – (H-E)] times 0.0092
3b	1.82E-08	[CDF – (H-E)] times 0.0023
7	4.97E-06	CDF – OK – (IS=F contribution) – (Class V contribution)
8	1.60E-08	Class V contribution from H-E

Class 1 Sequences – This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is determined from the Level 2 Release Category OK listed in Table 1, minus the EPRI Class 3a and 3b frequency, calculated below.

Class 2 Sequences – This group consists of all core damage accidents for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Release Category H-E, but only includes the contribution from failure of top event IS (containment isolation failure).

Class 3 Sequences – This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small (in excess of design allowable but <10La) or large (>100La).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.0092 [see Section 4.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0023 [see Section 4.3]

As described in Section 4.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions and H-E portion of Class 7).

CLASS_3A_FREQUENCY = 0.0092 * [CDF - Class 2 - Class 8 – (H-E part of Class 7)]
= 0.0092 * [CDF – (H-E)]
= 7.3E-8/yr

CLASS_3B_FREQUENCY = 0.0023 * [CDF - Class 2 - Class 8 – (H-E part of Class 7)]
=0.0023 * [CDF – (H-E)]
=1.8E-8/yr

For this analysis, the associated containment leakage for Class 3A is $10L_a$ and for Class 3B is $100L_a$. These assignments are consistent with the guidance provided in EPRI 1009325 Revision 2-A [27].

Class 4 Sequences – This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A test, this group is not evaluated further in the analysis.

Class 5 Sequences – This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components occurs. Because the failures are detected by Type C tests which are unaffected by the Type A test, this group is not evaluated further in this analysis.

Class 6 Sequences – This group is similar to Class 2. 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. Consistent with guidance provided in EPRI 1009325 Revision 2-A [27], this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences – This group consists of all core damage accidents in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined by subtracting the EPRI Class 1, 2, and 8 frequencies from total CDF.

Class 8 Sequences – This group consists of all core damage accidents in which containment bypass occurs. For this analysis, the frequency is determined from Release Category H-E, but only includes the contribution from Class V Level 1 core damage scenarios.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definitions of accident classes defined in EPRI 1009325 Revision 2-A [27]. Table 10 summarizes these accident frequencies by accident class for NMP2.

Table 10: Radionuclide Release Frequencies as a Function of Accident Class (Base Case)

Accident Classes (Containment Release Type)	Description	Frequency (per year)	
		Base Case	Base Case Plus Corrosion (1)
1	No Containment Failure	3.00E-6	3.00E-6
2	Large Isolation Failures (Failure to Close)	2.95E-7	2.95E-7
3a	Small Isolation Failures (liner breach)	7.29E-8	7.29E-8
3b	Large Isolation Failures (liner breach)	1.82E-8	1.83E-8
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA
7	Failures Induced by Phenomena (Early and Late)	4.97E-6	4.97E-6
8	Bypass (Interfacing System LOCA)	1.60E-8	1.60E-8
CDF	All CET end states	8.37E-6	8.37E-6

(1) Based on data developed in Section 4.4

5.2 Step 2 – Develop Plant-Specific Person-Rem Dose

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic and plant design differences compared to the reference plant, as described in Section 4.2, and summarized in Table 4. The results of applying these releases to the EPRI containment failure classification are as follows:

Table 11: NMP2 Dose Estimates for Population within 50 Miles

EPRI Class	Class Description	NMP2 Population Dose (1)	NMP2 Frequency (2)	Dose Rate (person-rem/yr) (3)
1	No Containment Failure (4)	3.75E+3	3.00E-6	1.13E-2
2	Containment Isolation Failure (5)	1.04E+6	2.95E-7	3.07E-1
3a	Small Pre-existing Leak (6)	3.75E+4	7.29E-8	2.74E-3
3b	Large Pre-existing Leak (7)	3.75E+5	1.82E-8	6.85E-3
7	Containment Failure – Severe Accident (8)	2.72E+5	4.97E-6	1.04
8	Containment Bypass (9)	1.04E+6	1.60E-8	1.67E-2
Totals		**NA**	**8.37E-6**	**1.39**

(1) Population dose taken from Table 4
(2) Frequency taken from Table 9
(3) Dose rate calculated by multiplying column 3 by column 4
(4) Population dose based on "no containment failure" APB 8 from NUREG/CR-4551
(5) Class 2 population dose based on NMP2 H-E set equal to APB 3 from NUREG/CR-4551
(6) Pre-existing small leak population dose is equal to 10 times EPRI Class 1 population dose
(7) Pre-existing large leak population dose is equal to 100 times EPRI Class 1 population dose
(8) Class 7 population dose and frequency are developed as follows

NMP2 Release Frequency		NUREG/CR-4551 APB	NMP2 Population Dose (person-rem)	Does Rate (person-rem/yr)
H-E (a)	1.31E-7	3	1.04E+6	1.37E-1
H-I + H-L	2.34E-7	6	7.98E+5	1.87E-1
M-E	2.48E-7	1	6.09E+5	1.51E-1
M-I + M-L	8.23E-7	2	3.82E+5	3.14E-1
L + LL	3.53E-6	9	7.18E+4	2.53E-1
Total	**4.97E-6**	**NA**	**2.10E+5(b)**	**1.04**

(a) Excludes EPRI Class 2 (H-E with IS=F) and 8 (H-E Class 5)
(b) Frequency-weighted population dose for EPRI class 7 obtained by dividing total population dose rate by the total release frequency

(9) Class 8 population dose based on NMP2 H-E set equal to APB 3 from NUREG/CR-4551

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [27] are provided in Table 11.

The above dose estimates, when combined with the results presented in Table 10, yield the NMP2 baseline mean consequence measures for each accident class. These results are presented in Table 12.

Table 12: NMP2 Annual Dose as a Function of Accident Class for Type A Test Required 3/10 Years

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	3.75E+3	3.00E-6	1.13E-2	3.00E-6	1.13E-2	-2.65E-7
2	Large Isolation Failures (Failure to Close)	1.04E+6	2.95E-7	3.07E-1	2.95E-7	3.07E-1	0.00
3a	Small Isolation Failures (liner breach)	3.75E+4	7.29E-8	2.74E-3	7.29E-8	2.74E-3	0.00
3b	Large Isolation Failures (liner breach)	3.75E+5	1.82E-8	6.85E-3	1.83E-8	6.87E-3	2.65E-5
7	Failures Induced by Phenomena (Early and Late)	2.10E+5	4.97E-6	1.04	4.97E-6	1.04	0.00
8	Bypass (Interfacing System LOCA)	1.04E+6	1.60E-8	1.67E-2	1.60E-8	1.67E-2	0.00
CDF	All Classes		8.37E-6	1.39	8.37E-6	1.39	2.62E-5

1) Only release Classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as $1L_a$ release magnitude consistent with the derivation of the Type A test non-detection failure probability. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.3 Step 3 – Evaluate Risk Impact of Extending Type A Test Interval From 10 to 15 Yrs

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to fifteen-years. To do this, an evaluation must first be made of the risk associated with the ten-year interval since the base case applies to a 3-year interval (i.e., a simplified representation of a 3-in-10 interval).

Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the methodology described in Section 4.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 13.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is higher by a factor of 5.0 compared to the 3-year interval value, as described in Section 4.3. The results for this calculation are presented in Table 14.

Table 13: NMP2 Annual Dose as a Function of Accident Class for Type A Test Required 1/10 Years

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	3.75E+3	2.79E-6	1.05E-2	2.79E-6	1.05E-2	-1.55E-6
2	Large Isolation Failures (Failure to Close)	1.04E+6	2.95E-7	3.07E-1	2.95E-7	3.07E-1	0.00
3a	Small Isolation Failures (liner breach)	3.75E+4	2.43E-7	9.12E-3	2.43E-7	9.12E-3	0.00
3b	Large Isolation Failures (liner breach)	3.75E+5	6.07E-8	2.28E-2	6.11E-8	2.30E-2	1.55E-4
7	Failures Induced by Phenomena (Early and Late)	2.10E+5	4.97E-6	1.04	4.97E-6	1.04	0.00
8	Bypass (Interfacing System LOCA)	1.04E+6	1.60E-8	1.67E-2	1.60E-8	1.67E-2	0.00
CDF	All Classes		8.37E-6	1.41	8.37E-6	1.41	1.53E-4

1) Only release classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as $1L_a$ release magnitude consistent with the derivation of the Type A test non-detection failure probability. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 14: NMP2 Annual Dose as a Function of Accident Class for Type A Test Required 1/15 Years

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	EPRI Methodology		EPRI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr(1)
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure (2)	3.75E+3	2.64E-6	9.90E-3	2.64E-6	9.90E-3	-3.57E-6
2	Large Isolation Failures (Failure to Close)	1.04E+6	2.95E-7	3.07E-1	2.95E-7	3.07E-1	0.00
3a	Small Isolation Failures (liner breach)	3.75E+4	3.65E-7	1.37E-2	3.65E-7	1.37E-2	0.00
3b	Large Isolation Failures (liner breach)	3.75E+5	9.12E-8	3.42E-2	9.21E-8	3.46E-2	3.57E-4
4	Small Isolation Failures (Failure to seal type B)	2.10E+5	4.97E-6	1.04	4.97E-6	1.04	0.00
8	Bypass (Interfacing System LOCA)	1.04E+6	1.60E-8	1.67E-2	1.60E-8	1.67E-2	0.00
CDF	All Classes		8.37E-6	1.42	8.37E-6	1.42	3.54E-4

1) Only release classes 1 and 3b are affected by the corrosion analysis.
2) Characterized as $1L_a$ release magnitude consistent with the derivation of the Type A test non-detection failure probability. Release classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.4 Step 4 – Determine Change in Risk in Terms of LERF

The risk increase associated with extending the Type A test interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the EPRI guidance, 100% of the Class 3b contribution would be considered LERF.

Regulatory Guide 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 core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the Type A test interval does not impact CDF, the relevant metric is LERF.

For NMP2, 100% of the frequency of Class 3b sequences can be used as a conservative first-order estimate to approximate the potential increase in LERF from the Type A test interval extension (consistent with the EPRI guidance methodology). Based on a ten-year test interval from Table 13 the Class 3b frequency is 6.07E-8/yr; and, based on a fifteen-year test interval from Table 14, it is 9.12E-8/yr. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the Type A test interval from 3 to 15 years is 7.3E-8/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is 3.1E-8/yr. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF is below the threshold criteria for a very small change.

5.5 Step 5 – Determine Impact on Conditional Containment Failure Probability

Another parameter that the NRC guidance in RG 1.174 states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the Type A test interval on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis. In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP is calculated as a basis for demonstrating that the proposed change is consistent with the defense-in-depth philosophy. The change in CCFP is calculated by using the method specified in the EPRI 1009325 Revision 2-A [27] as follows:

CCFP = [1 - (Class 1 frequency + Class 3a frequency) / CDF] * 100%

$CCFP_3 = 63.27\%$

$CCFP_{10} = 63.78\%$

$CCFP_{15} = 64.14\%$

$$\Delta CCFP = CCFP_{15} - CCFP_3 = 0.87\%$$

$$\Delta CCFP = CCFP_{15} - CCFP_{10} = 0.36\%$$

The change in CCFP of less than 1% by extending the Type A test interval to 15 years from the original 3-in-10 year requirement is judged to be insignificant.

5.6 Summary of Results

The results from this Type A test interval extension risk assessment for NMP2 are summarized in Table 15.

Table 15: Summary of NMP2 Results for the Type A Test Interval Change

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		CDF	Per-Rem/Yr	CDF	Per-Rem/Yr	CDF	Per-Rem/Yr
1	3.75E+3	3.00E-6	1.13E-2	2.79E-6	1.05E-2	2.64E-6	9.90E-3
2	1.04E+6	2.95E-7	3.07E-1	2.95E-7	3.07E-1	2.95E-7	3.07E-1
3a	3.75E+4	7.29E-8	2.74E-3	2.43E-7	9.12E-3	3.65E-7	1.37E-2
3b	3.75E+5	1.82E-8	6.85E-3	6.07E-8	2.28E-2	9.12E-8	3.42E-2
7	2.10E+5	4.97E-6	1.04	4.97E-6	1.04	4.97E-6	1.04
8	1.04E+6	1.60E-8	1.67E-2	1.60E-8	1.67E-2	1.60E-8	1.67E-2
Total		8.37E-6	1.39	8.37E-6	1.41	8.37E-6	1.42
ILRT Dose Rate from 3a and 3b		9.58E-3		3.19E-2		4.79E-2	
Delta Total Dose Rate	From 3 yr	N/A		2.15E-2		3.70E-2	
	From 10 yr	N/A		N/A		1.54E-2	
% change in dose rate from base	From 3 yr	N/A		1.55%		2.67%	
	From 10 yr	N/A		N/A		1.10%	
3b Frequency (LERF)		1.82E-8		6.07E-8		9.12E-8	
Delta LERF	From 3 yr	N/A		4.25E-8		7.29E-8	
	From 10 yr	N/A		N/A		3.05E-8	
CCFP %		63.27%		63.78%		64.14%	
Delta CCFP %	From 3 yr	N/A		0.51%		0.87%	
	From 10 yr	N/A		N/A		0.36%	

6 Sensitivities

6.1 Sensitivity of Corrosion Impact Assumptions

The results in Tables 12, 13 and 14 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the Type A test interval extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the cylinder and dome and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 16. In every case the impact from including the corrosion effects is very minimal. Even the upper bound estimates with very conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 2.8E-8/yr. The results indicate that even with very conservative assumptions, the conclusions from the base analysis would not change.

Table 16: Steel Liner Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	**Containment Breach (Step 4 in the corrosion analysis)**	**Visual Inspection & Non-Visual Flaws (Step 5 in the corrosion analysis)**	**Increase in Class 3b Frequency (LERF) for Type A Test Extension 3 to 15 years (per Rx-yr)**	
			Total Increase	**Increase Due to Corrosion**
Base Case Doubles every 5 yrs	Base Case (1% Cylinder 0.1% Basemat)	Base Case (10% Cylinder 100% Basemat)	7.38E-8	8.81E-10
Doubles every 2 yrs	Base	Base	7.49E-8	1.97E-9
Doubles every 10 yrs	Base	Base	7.37E-8	7.25E-10
Base	Base	*15%*	7.42E-8	1.24E-9
Base	Base	*5%*	7.35E-8	5.32E-10
Base	10% Cylinder 1% Basemat	Base	8.18E-8	8.87E-9
Base	0.1% Cylinder 0.01% Basemat	Base	7.30E-8	8.87E-11
Lower Bound				
Doubles every 10 yrs	0.1% Cylinder 0.01% Basemat	5% 1%	7.30E-8	2.91E-11
Upper Bound				
Doubles every 2 yrs	10% Cylinder 1% Basemat	15% 100%	1.00E-7	2.75E-8

6.2 Sensitivity to Class 3B Contribution to LERF

The Class 3b frequency for the base case of a three in ten-year Type A test interval is 1.82E-8/yr [Table 12]. Extending the interval to one in ten years results in a frequency of 6.07E-8/yr [Table 13]. Extending it to one in fifteen years results in a frequency of 9.12E-8/yr [Table 14], which is an increase of 7.3E-8/yr. If 100% of the Class 3b sequences are assumed to have potential releases large enough for LERF, then the increase in LERF due to extending the interval from three in ten to one in fifteen years is below the RG 1.174 threshold for very small changes in LERF of 1E-7/yr.

6.3 Potential Impact from External Events

In the NMP2 IPEEE, the dominant risk contributor from external events was found to be from fire and seismic events. Other potential contributors such as transportation and high winds were found to be within acceptable limits. The fire and seismic IPEEE analyses were subsequently incorporated into the NMP2 PRA model; their CDF and LERF contributions in the NMP2 PRA are provided below:

Hazard	CDF	LERF
Fire	3.0E-6	5.1E-7
Seismic	1.8E-7	1.4E-7
Total	3.1E-6	6.6E-7

Although the fire and seismic IPEEE analyses were incorporated into the NMP2 PRA model, these hazard analyses and models have not been updated since the original IPEEE although the PRA model itself is maintained current with plant design and operation. Thus, the change in LERF from extending the Type A test interval is conservatively estimated using the total external event CDF for fires and seismic. Table 17 shows the results of these calculations.

Table 17: NMP2 Estimated Total LERF Including External Events Impact

Case	3b Frequency (3-per-10 year test)	3b Frequency (1-per-10 year test)	3b Frequency (1-per-15 year test)	LERF Increase (3-per-10 to 1-per-10)	LERF Increase (3-per-10 to 1-per-15)	LERF Increase (1-per-10 to 1-per-15)
External Event Contribution	7.13E-9	2.35E-8	3.57E-8	1.64E-8	2.85E-8	1.21E-8
Internal Event Contribution	1.82E-8	6.07E-8	9.12E-8	4.25E-8	7.29E-8	3.05E-8
Combined (Internal+External)	2.54E-8	8.43E-8	1.27E-7	5.89E-8	1.01E-7	4.26E-8

An increase in LERF of 1.01E-7 is within the range of 1E-07/yr to 1E-06/yr (Region II of the RG 1.174 LERF acceptability curve). This is considered an acceptable small increase if total LERF is less than 1E-5/yr.

7 Conclusion

Based on the results from Section 5 and the sensitivity calculations presented in Section 6, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A test interval to fifteen years:

- Regulatory Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test frequency from three in ten years to one in fifteen years is conservatively estimated as 7.3E-8/yr using the EPRI guidance as written. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Regulatory Guide 1.174.

- Regulatory Guide 1.174 [4] also states that when the calculated increase in LERF is in the range of 1.0E-06 per reactor year to 1.0E-07 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. An additional assessment of the impact from external events was also made indicating a combined LERF increase of 1.01E-7. In this case, the total LERF (combined internal and external events) for NMP2 is well below the RG 1.174 acceptance criteria for total LERF of 1.0E-05.

- For the change in Type A test frequency to once-per-fifteen-years, the calculated increase to the total 50-mile population dose for those accident sequences influenced by Type A testing is 3.7E-2 person-rem/yr. EPRI 1009325 Revision 2-A [27] states that a very small population dose increase is defined as ≤ 1.0 person-rem per year or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended Type A test interval. Moreover, the risk impact when compared to other severe accident risks is negligible.

- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen year interval is 0.87%. EPRI 1009325 Revision 2-A [27] states that increases in CCFP of ≤ 1.5 percentage points is very small. Therefore this increase is judged to be very small.

The above findings of the NMP2-specific risk assessment confirm the general findings of previous studies (NUREG-1493 and EPRI) that increasing the Type A test interval to 15 years results in a very small change to the NMP2 risk profile.

8 References

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