FENOC®

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March 7, 2018
L-18-046

10 CFR 50.90

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
License Amendment Request to Extend Containment Leakage Test Frequency

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) hereby requests an amendment to the facility operating license for the Perry Nuclear Power Plant (PNPP). The proposed license amendment would revise Technical Specification 5.5.12, "Containment Leakage Rate Testing Program," to follow the guidance provided in Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J," with conditions and limitations specified in NEI 94-01, Revision 2-A (having the same title as Revision 3-A), instead of Regulatory Guide 1.163, "Performance Based Containment Leak Test Program." The proposed license amendment would also revise Technical Specification 5.5.12 by deleting two of the four listed exceptions to program guidelines, since they are no longer necessary.

The FENOC evaluation of the proposed changes is enclosed. Approval of the proposed amendment is requested by March 9, 2019 to support related testing activities to be conducted during the next PNPP refueling outage (1R17), which is currently scheduled to begin in March 2019. Once approved, the amendment shall be implemented within 30 days of receipt.

There are no regulatory commitments contained in this submittal. If there are any questions or additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 7, 2018.

Sincerely,

David B. Hamilton

Enclosure:
FENOC Evaluation of the Proposed Amendment

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager
Branch Chief, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

Subject: License Amendment Request to Revise Technical Specification 5.5.12 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies

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Attachments

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the facility operating license for Perry Nuclear Power Plant (PNPP).

The proposed license amendment would revise Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to follow guidance developed by the Nuclear Energy Institute (NEI) in topical report (TR), NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, with conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, with the same title (Accession Numbers ML12221A202 and ML100620847, respectively). The proposed license amendment would also revise TS 5.5.12 by deleting two of the four listed exceptions to program guidelines, since they are no longer necessary.

The purpose of the NEI 94-01 guidance is to assist licensees in implementing Option B, "Performance Based Requirements," of 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Nuclear Power Reactors." Revision 2-A of NEI 94-01 added guidance for extending Type A surveillance test intervals beyond ten years, and Revision 3-A of NEI 94-01 added guidance for extending Type C surveillance test intervals beyond sixty months.

This amendment will allow extension of the Type A test interval up to one test in 15-years and extension of the Type C test interval up to 75-months, based on acceptable performance history as defined in NEI 94-01, Revision 3-A.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

Primary Containment

The primary containment is a pressure retaining structure composed of a free-standing steel cylinder with an ellipsoidal dome secured to a steel lined reinforced concrete foundation mat. The primary containment is surrounded by a reinforced concrete structure (the shield building). The primary containment and shield building are tied together by the annulus concrete. These structures are supported by a common foundation mat. PNPP Updated Safety Analysis Report (USAR) Figure 3.8-1 provides a simplified cross-section of the structures.

The primary containment is designed to contain radioactive material that might be released from the reactor coolant system following a loss-of-coolant accident (LOCA). The primary containment ensures a high degree of leak tightness during normal operating and accident conditions. It is designed for a maximum internal pressure of

15 pounds per square inch gauge (psig) at accident conditions. The maximum design leakage rate is 0.2 percent by weight of the contained atmosphere in 24 hours at 7.80 psig internal pressure.

Steel test channels are provided along welds of the steel liner of the foundation mat so that local leak testing of welds can be performed. The channels are segmented by area. One plug fitting is provided for each area through any covering material including concrete.

There are two personnel air locks that permit passage from the outside of the shield building into the primary containment. PNPP USAR Figure 3.8-4 provides a drawing of an airlock. The personnel air locks are welded steel assemblies with double doors, each equipped with double gaskets, and connections for leak rate testing.

For equipment access into the primary containment during plant outages, there is a removable, bolted equipment hatch. PNPP USAR Figure 3.8-5 provides a drawing of the equipment hatch. The flanged joint between the hatch and cover is designed to accommodate double seals. Periodic leak testing of the hatch is accomplished by pressurizing the space between the seals.

There are two 24-inch nominal diameter vacuum relief lines provided to obtain the vacuum relief cross-sectional area required to prevent a postulated negative pressure inside containment from exceeding the design value of 0.8 psi. Two additional 24-inch relief lines are provided for redundancy. Each relief line has a simple check valve inside containment. This check valve serves as both the vacuum breaker device, and the inner isolation valve for vacuum relief line. Outside containment, each vacuum relief line has a 24-inch nominal diameter motor-operated butterfly valve to serve as the outer isolation valve.

Mechanical and electrical lines that enter or exit the primary containment pass through penetrations located in both the primary containment and shield building walls. Typical mechanical and electrical penetrations are shown in PNPP USAR Figures 3.8-6 and 3.8-7. Penetrations are of the double barrier type and are designed to provide an annular air space that can be pressurized for leak testing.

To minimize the amount of primary containment leakage following a design basis accident, mechanical systems that pass through both the primary containment and shield building are typically provided with redundant containment isolation barriers one inside of the primary containment and one outside of the shield building, or some other acceptable configuration such as a closed system outside of containment.

USAR Table 6.2-32, “Containment Isolation Valve Summary,” provides a listing of the primary containment penetrations and their associated primary containment isolation barriers. The majority of the penetrations bypass the secondary containment and any leakage would not be processed by the annulus exhaust gas treatment (AEGT) system.

Specific leakage limits for these penetrations have been established and are consistent with the existing accident analysis. USAR Table 6.2-33, "Potential Secondary Containment Bypass Leakage Paths," provides the potential leakage paths that could bypass the secondary containment and the AEGT system.

Secondary Containment

The shield building is a reinforced concrete structure consisting of a flat foundation mat, a cylindrical wall and a shallow dome. The functions of the shield building include but are not limited to:

- Provides weather and exterior missile protection for the primary containment.
- Provides a relatively leak tight structure that collects fission product leakage from the primary containment during and following a postulated design basis accident and delays it until processing it through the annulus exhaust gas treatment (AEGT) system to minimize the escape of radioactive particles to the environment, by maintaining the annulus air space at a slight negative pressure.

The secondary containment consists of the volume (or annulus) between the shield building and the primary containment. This area is five feet wide, and has numerous systems that pass through both the primary containment and shield building walls as shown by PNPP USAR Figure 6.2-60 (Sheets 1 through 4). There are two doors, which are normally locked, that permit access to the annulus. Additionally, there is a large opening in the shield building to provide access to the primary containment equipment hatch. This opening is shielded by eight large removable reinforced concrete beams (called shield beams or shield blocks) as shown in PNPP USAR Figure 3.8-5, which are provided with seals to minimize air leakage. To establish secondary containment operability, the activities include but are not limited to ensuring the primary containment equipment hatch and the annulus access doors are closed, ensuring the shield beams (with seals) are installed, and the AEGT system is operable.

The AEGT system functions to maintain a negative pressure differential between the annulus and outside of the shield building. The AEGT system processes the air in the annular space between the shield building and the primary containment to limit the release to the environment of radioisotopes that may leak from the primary containment under accident conditions. Sources of leakage into the annulus (secondary containment) include but are not limited to, primary containment shell leakage and primary containment airlock outer door seal leakage. The AEGT system is a recirculation type system with split flow. Some of the filtered air extracted from the annulus space is recirculated and some is discharged to the unit vent. The system has two subsystems; each subsystem includes a high efficiency particulate (HEPA) pre-filter, a charcoal adsorber, and a HEPA post-filter.

2.2 Current Technical Specification Requirements

On September 9, 1997, the NRC approved TS Amendment 86 (Accession Number ML021840313) authorizing the implementation of 10 CFR 50, Appendix J, Option B for Type A, B, and C tests. TS 5.5.12 currently states, in part that:

> A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:
>
> - BN-TOP-1 methodology may be used for Type A tests.
> - The corrections to NEI 94-01 which are identified on the Errata Sheet attached to the NEI letter, "Appendix J Workshop Questions and Answers," dated March 19, 1996 are considered an integral part of NEI 94-01.
> - The containment isolation check valves in the Feedwater penetrations are tested per the INSERVICE TESTING PROGRAM.
> - The provisions of NEI 94-01, Section 9.2.3 are revised to include the following exception: The first Type A test performed after the Type A test completed on July 1, 1994 shall be completed no later than June 29, 2009.

Adoption of the Option B performance-based containment leakage rate testing program altered the frequency of measuring primary containment leakage in Types A, B, and C tests but did not alter the basic method by which Appendix J leakage testing is performed. The test frequency is based on an evaluation of the "as found" leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will not be exceeded.

2.3 Reason for the Proposed Change

The guidance in NEI 94-01 was developed to assist licensees in the implementation of Option B to 10 CFR 50, Appendix J, "Leakage Rate Testing of Containment of Light Water Cooled Nuclear Power Plants." It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights.

Revision 2-A of NEI 94-01 added guidance for extending Type A Integrated Leak Rate Test (ILRT) surveillance intervals beyond ten years, and Revision 3-A of NEI 94-01 added guidance for extending Type C Local Leak Rate Test (LLRT) surveillance intervals beyond sixty months.

The proposed amendment would change TS 5.5.12 to state that the Primary Containment Leakage Rate Testing Program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, instead of the guidelines of Regulatory Guide 1.163, to allow extension of the Type A test interval up to one test in 15-years and extension of the Type C test interval up to 75-months.

The TS 5.5.12 exception to use certain corrections to NEI 94-01, Revision 0, (Reference 1), is to be deleted since this revision of NEI 94-01 will no longer be in use. The TS 5.5.12 exception regarding a Type A test to be completed no later than June 29, 2009, is to be deleted since the test has already been performed. Deletion of these exceptions is an administrative action that has no effect on any component and no impact on how the unit is operated.

Another exception permits use of the BN-TOP-1 testing methodology for Type A tests. The BN-TOP-1 testing methodology exception will be retained in TS 5.5.12, since this methodology continues to provide acceptable results when this option is used to perform Type A tests.

The final exception requires that containment isolation check valves in the Feedwater penetrations are to be tested per the Inservice Testing Program. This exception will be retained since the exception documents how testing will be performed.

2.4 Description of the Proposed Change

The proposed license amendment would revise the TS 5.5.12 wording by:

- Replacing the reference to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with a reference to NEI Topical Report NEI 94-01, Revisions 3-A, with conditions and limitations in NEI 94-01, Revision 2-A,
- Deleting the exception to use the corrections to NEI 94-01, which are identified on the Errata Sheet attached to the NEI letter, "Appendix J Workshop Questions and Answers," dated March 19, 1996, and
- Deleting the exception regarding the performance of the first Type A test performed after the Type A test completed on July 1, 1994 shall be completed no later than June 29, 2009.

Conditions and limitations in NEI 94-01, Revision 2-A, are presented in Section 4.0 of the June 25, 2008 NRC safety evaluation that has been incorporated into NEI 94-01, Revision 2-A.

Attachment 1 provides a copy of TS pages 5.0-15 and 5.0-15a marked to show the proposed changes. Text to be deleted is marked with a line through the letters, and text to be added is shown in a box.

3.0 TECHNICAL EVALUATION

The following technical evaluation subsections provide supporting information used in evaluating and justifying the proposed TS change, and addressing NRC concerns discussed in similar applications. The sections provide information regarding the impact of ECCS overpressure on the containment structure, historical Appendix J requirements for PNPP, detailed historical integrated leakage rate testing results, plant specific risk and non-risk assessments for extended test intervals, plant operating experience, containment modifications, planned license renewal application, and NEI 94-01, Revision 2-A, and Revision 3-A, limitations and conditions.

3.1 Emergency Core Cooling System (ECCS) Net Positive Suction Head (NPSH) Analysis

Regulatory Guide 1.1, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," November 1970, prohibits design reliance on pressure and/or temperature transients expected during a LOCA for assuring adequate NPSH. The guidelines are applicable to the high pressure core spray (HPCS), low pressure core spray (LPCS) and the low pressure core injection pumps.

The BWR design conservatively assumes 0 psig containment pressure and maximum expected temperatures of the pumped fluids. Thus, no reliance is placed on pressure or temperature transients to assure adequate NPSH.

3.2 Justification for the Technical Specification Change

3.2.1 Chronology of 10 CFR 50, Appendix J, Testing Requirements

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 or associated bases. 10 CFR 50, Appendix J also ensures that periodic surveillance of reactor containment penetrations and isolation valves are performed so that proper maintenance and repairs are made during the service life of the containment and those 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 loss of coolant accident.

Appendix J identifies three types of required tests: 1) Type A tests, intended to measure the primary containment overall integrated leakage rate (also referred to as ILRT); 2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries for primary containment penetrations, and; 3) Type C tests, intended to measure containment isolation valve leakage rates. Type A

tests focus on verifying the leakage integrity of a passive containment structure. Type B and C tests (also referred to as local leakage rate tests or LLRTs) focus on assuring that containment penetrations are essentially leak tight. Types 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, 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, Regulatory Guide 1.163 was issued. The Regulatory Guide endorsed NEI 94-01, Revision 0, with certain modifications and additions. Option B, in concert with Regulatory Guide 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (that is, 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.

In 2008, NEI 94-01, Revision 2-A, was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, subject to the limitations and conditions noted in Section 4.0 of the June 25, 2008 NRC safety evaluation (Accession Number ML081140105) for NEI 94-01, Revision 2-A. NEI 94-01, Revision 2-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights. The approach includes provisions for extending Type A ILRT intervals to up to 15 years and incorporates the regulatory positions stated in Regulatory Guide 1.163.

In 2012, NEI 94-01, Revision 3-A, was issued. The NRC staff determined that this document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J as modified by the conditions and limitations summarized in Section 4.0 of the June 8, 2012 NRC safety evaluation (Accession Number ML121030286) for NEI 94-01, Revision 3-A. NEI 94-01, Revision 3-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies.

Justification for extending test intervals is based on the performance history and risk insights. Extensions of Type B and Type C test intervals are allowed based upon completion of two consecutive periodic as-found tests where the results of each test are within a licensee's allowable administrative limits. Intervals may be increased from 30 months up to a maximum of 120 months for Type B tests (except for containment airlocks) and up to a maximum of 75 months for Type C tests. If a licensee considers extended test intervals of greater than 60 months for Type B or Type C tested components, the review should include the additional considerations of as-found tests, schedule and review as described in NEI 94-01, Revision 3-A, Section 11.3.2, "Programmatic Controls."

NEI 94-01, Revision 3-A, has been endorsed as providing methods acceptable to the NRC staff for complying with the provisions of 10 CFR Part 50, Appendix J, Option B, subject to the regulatory positions in Regulatory Guide 1.163 and conditions and limitations summarized in Section 4 of the NRC safety evaluations dated June 25, 2008 and June 8, 2012.

The allowed frequency for Type A testing as documented in NEI 94-01 is based, in part, upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," published in September of 1995. The evaluation documented in NUREG-1493 includes a study of the dependence of reactor accident risks on containment leak tightness for different containment types, including a pressure suppression containment similar to the PNPP containment structure. NUREG-1493 concludes in Section 10.1.2 that reducing the frequency of Type A tests (ILRTs) from three tests per ten-years to one test per 20-years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Types B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements. Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, NUREG-1493 concludes that increasing the interval between ILRTs is possible with minimal impact on public risk.

The NRC has provided guidance concerning the use of test interval extensions in the deferral of ILRTs beyond the 15-year interval, in the June 25, 2008 NRC safety evaluation for NEI 94-01, Revision 2-A. Section 3.1.1.2, "Deferral of Tests Beyond The 15-Year Interval," of the safety evaluation states in part:

> Section 9.2.3, NEI TR 94-01, Revision 2, states, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 15 years based on acceptable performance history." However, Section 9.1 states that the "required surveillance intervals for recommended

> Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes." The NRC staff believes that extensions of the performance-based Type A test interval beyond the required 15 years should be infrequent and used only for compelling reasons. Therefore, if a licensee wants to use the provisions of Section 9.1 in TR NEI 94-01, Revision 2, the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists.

The NRC has also provided the following concerning the extension of ILRT intervals to 15 years in NEI 94-01, Revision 3-A, NRC safety evaluation, Section 4.0, Condition 2, which states, in part:

> The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time.

NEI 94-01, Revision 3-A, Section 10.1 , "Introduction," concerning the use of test interval extensions in the deferral of Type B and Type C LLRTs, based on performance, states in part, that:

> Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals of up to 120 months for the recommended surveillance frequency for Type B testing and up to 75 months for Type C testing given in this section may be extended by up to 25 percent of the test interval, not to exceed nine months.
>
> Notes: For routine scheduling of tests at intervals over 60 months, refer to the additional requirements of Section 11.3.2.
>
> Extensions of up to nine months (total maximum interval of 84 months for Type C tests) are permissible only for non-routine emergent conditions. This provision (nine-month extension) does not apply to valves that are restricted and/or limited to 30 month intervals in Section 10.2 (such as BWR MSIVs) or to valves held to the base interval (30 months) due to unsatisfactory LLRT performance.

3.2.2 Integrated Leak Rate Test (ILRT) History

On September 9, 1997, the NRC approved Amendment 86 to PNPP TS. The amendment added the primary containment leakage rate testing program (TS 5.5.12) that implements 10 CFR 50, Appendix J, Option B in accordance with the guidelines

contained in Regulatory Guide 1.163, as modified by certain exceptions. Regulatory Guide 1.163 endorsed NEI 94-01, Revision 0, as an acceptable means of demonstrating compliance with the 10 CFR 50, Appendix J, Option B, subject to certain conditions. Since the TS was revised to implement Option B in 1997, the performance leakage rates have been calculated in accordance with NEI 94-01, Revision 0, Section 9.1.1, "Performance Criteria," for Type A testing. Table 3.2.2-1 below lists the past periodic Type A test results.

Table 3.2.2-1 PNPP Type A Test History

Test Date	Leakage 95% Upper Confidence Level, wt. %/day (Note 1)	Total Leakage As-Found, wt. %/day (Note 2)	Total Leakage As-Left, wt. %/day (Note 3)	Accident Pressure, Pa psig (Note 4)	Acceptance Limit As-Found / As-Left (Note 5)
1985	0.0611	Note 6	0.0639	11.31	0.20 / 0.15
1989	0.029	Note 6	0.101	11.31	0.20 / 0.15
1993	0.1007	Note 6	0.1121	11.31	0.20 / 0.15
1994 Note 7	0.1063	Note 6	0.1122	11.31	0.20 / 0.15
2009	0.1195	0.1382	0.1145	7.80	0.20 / 0.15

Notes:

1. ILRT leakage values determined by the end of test unadjusted 95 percent upper confidence limit. No Type B and C penalties or sump level change penalties were assigned or included in this leak rate test value. "%" is used as an abbreviation for "percent," and "wt. %/day" is used as an abbreviation for "weight percent per day."
2. ILRT as-found leak rate test data contains 95 percent upper confidence limit and all penalties assigned including leakage value adjustments from Type B and C component repairs performed prior to the ILRT test and sump level changes during the ILRT test.
3. ILRT as-left leak rate test data contains 95 percent upper confidence limit and all adjusted penalties (for example, sump level changes and isolated volumes)
4. The NRC approved TS Amendment No. 57 to PNPP Facility Operating License NPF-58 on March 23, 1994 (Accession No. ML021830439) which changed the containment accident pressure (Pa) from 11.31 psig to 7.80 psig.

5. The maximum allowable primary containment leakage rate (L_a) and the maximum allowable primary containment leakage rate during the first unit startup following testing (0.75 L_a), shall be 0.20 percent and 0.15 percent of primary containment air weight per day at the peak containment pressure (Pa), respectively. The maximum primary containment leakage rate was less than or equal to 1.0 L_a for the as-found ILRT testing and less than or equal to 0.75 L_a for the as-left testing for startup.
6. Test performed prior to 10 CFR 50, Appendix J, Option B as-found reporting requirements.
7. On July 8, 1993, the results of the Reactor Containment Building Integrated Leak Rate Test (ILRT) was categorized as a failure based on adjustment of the Type A test results with the "as-found" data from Type B and C penetration tests. The failure resulted in the ILRT schedule being accelerated to be performed every refueling outage until two consecutive ILRTs met the acceptance criteria. Therefore, an ILRT was conducted from June 29 through July 1, 1994 during the fourth refueling outage.

   However, correspondence from the Director of the Office of Nuclear Reactor Regulation to Duke Power dated July 27, 1989, clarified that 10 CFR 50, Appendix J did not require licensees to correct Type A test results for certain Type B and C leakage. The 1993 ILRT as-found results did not consider this guidance and was determined to be overly conservative. The test results were re-evaluated, the 1993 ILRT was not considered a failure, and accelerated testing was not required. The test results were communicated in a letter to the NRC dated February 3, 1995.

For purposes of determining the extended ILRT interval, the performance leakage rate was calculated for the 2009 ILRT performance. The 2009 ILRT leakage rates were calculated using the Mass Point Analysis equations described in American National Standards Institute, American Nuclear Society (ANSI/ANS) Standard 56.8-1994, "Containment System Leakage Testing Requirements." During the 2009 ILRT, the minimum pathway leak rate (MNPLR) for Type B and Type C pathways that were not exposed to the ILRT pressure was 0.024 weight percent per day. The pathways which contributed to the MNPLR are as follows:

Table 3.2.2-2 2009 ILRT Pretest Leakage Penalties for Non-Vented or Isolated Penetrations

Penetration No.	Inside Barrier	Leakage (SCCM)	Outside Barrier	Leakage (SCCM)	MNPLR (SCCM)
P104	N/A	N/A	1E51-F019	2	2
P106	1E51-F068	435	1E51-F040	134	134
P107	N/A	N/A	1E12-F055A	69	70

Table 3.2.2-2 2009 ILRT Pretest Leakage Penalties for Non-Vented or Isolated Penetrations

Penetration No.	Inside Barrier	Leakage (SCCM)	Outside Barrier	Leakage (SCCM)	MNPLR (SCCM)
			1E12-F025A 1E21-F018		
			1N27-F751 1N27-D027	1	
P108	1P11-F545	2.4	1P11-F060	2	2
P109	1E61-D017	N/A	1E61-D003	1	1
	Blank Flange	N/A			
P111	1P11-F090	85	1P11-F080	2.6	2.6
P112	1E12-F006	44.7	1E21-F005	61.2	44.7
P113	1E12-F028A	17.5	1E12-F027A	80.7	42.5
	1E12-F037A	20			
	1E12-F042A	5			
P115	1E12-F102	80	N/A	N/A	80
P116	1P57-F524B	97.6	1P57-F015B	80.6	80.6
P118	1E12-F558A	2	1E12-F073A	2	2
P119	1E61-D014	N/A	1E61-D015	1	1
P120	1E61-D016	N/A	1E61-D001	20	20
	Blank Flange	N/A			
P121	N/A	N/A	1E12-F053A	4950	4950
P123	1E51-F066	2	1E51-F013	1	1
			1E12-F023		
P131	1G33-F001	20	1G33-F004	43	20
P132	1G33-F040	24.8	1G33-F039	43	24.8
P203	1G41-F522	30	1G41-F100	3.4	3.4
P204	1C11-F122	2	1C11-F083	20	2
P205	Blank Flange	40	1F42-F003	N/A	40
			1F42-N016		
			IFTS Bellows		
P301	1G41-F140	156.3	1G41-F145	248.5	158.3

Table 3.2.2-2 2009 ILRT Pretest Leakage Penalties for Non-Vented or Isolated Penetrations

Penetration No.	Inside Barrier	Leakage (SCCM)	Outside Barrier	Leakage (SCCM)	MNPLR (SCCM)
	1G41-F801	2			
P304	1P57-F524A	13.2	1P57-F015A	2	2
P309	1P22-F577	196.5	1P22-F010	5.3	5.3
P310	1P43-F721	2	1P43-F055	2	2
P311	1P43-F215	2	1P43-F140	2	2
	1P43-F851	2			
V314	1M14-F085	N/A	1M14-F090	42.8	42.8
	1M14-F200				
P315	1C41-F520	21.7	1C41-F518	2	2
P317	1E61-F549	N/A	1E61-D007	1.1	1.1
	Pipe Cap	N/A			
P317	1E61-F550	N/A	1E61-D006	1.3	1.3
	Pipe Cap	N/A			
P318	N/A	N/A	1M51-F210B 1P87-F074	7.1	7.1
P318	N/A	N/A	1M51-F220B	6.6	6.6
P318	N/A	N/A	1M51-F230B 1P87-F077	2	2
P318	N/A	N/A	1M51-F240B 1P87-F071	2	2
P318	N/A	N/A	1M51-F250B 1P87-F065	2	2
P319	1E61-F551	N/A	1E61-D005	2	2
	Pipe Cap	N/A			
P319	1E61-F552	N/A	1E61-D004	2	2
	Pipe Cap	N/A			
P404	1P50-F539	2	1P50-F060	21.6	2
P405	1P50-F140	2	1P50-F150	129.5	4

Table 3.2.2-2 2009 ILRT Pretest Leakage Penalties for Non-Vented or Isolated Penetrations

Penetration No.	Inside Barrier	Leakage (SCCM)	Outside Barrier	Leakage (SCCM)	MNPLR (SCCM)
	1P50-F606	2			
P406	1P54-F727	55	1P54-F726	3.7	3.7
P408	N/A	N/A	1E12-F021	27.5	27.5
P409	N/A	N/A	1E22-F012 1E22-F023 1E22-F035	128	128
P413	1P87-F049	2	1P87-F055	2	2
	1P87-F277	2			
P413	1P87-F046	2	1P87-F052	2	2
P414	N/A	N/A	1E12-F053B	1020.5	1020.5
P417	1G61-F075	2	1G61-F080	2	2
	1G61-F655	2			
P418	1G61-F165	2	1G61-F170	2	2
	1G61-F657	2			
P419	1G33-F053	37.6	1G33-F054	42.8	37.6
P420	1G50-F272	1	1G50-F277	1	1
	1G50-F823	1			
P421	1E12-F009 1E12-F550	3210	1E12-F008	3110	3110
P422	1E51-F063	N/A	1E51-F064	41.3	20.7
	1E51-F076	N/A			
P423	1B21-F016	N/A	1B21-F019	86.5	44
P424	1G33-F028	32.2	1G33-F034	64.4	32.2
	1G33-F646	2			
P425	N/A	N/A	1M51-F210A	24.9	24.9
P425	N/A	N/A	1M51-F220A	33.8	33.8
P425	N/A	N/A	1M51-F230A	63.6	63.6
P425	N/A	N/A	1M51-F240A	4.9	4.9
P425	N/A	N/A	1M51-F250A	8.3	8.3
P429	1P87-F264	3.1	1P87-F083	2	2

Table 3.2.2-2 2009 ILRT Pretest Leakage Penalties for Non-Vented or Isolated Penetrations					
Penetration No.	Inside Barrier	Leakage (SCCM)	Outside Barrier	Leakage (SCCM)	MNPLR (SCCM)
P429	N/A	N/A	1E12-F005 1E12-F025B 1E12-F025C 1E12-F055B	126.1	126.1
P431	1E12-F558B	2	1E12-F073B	2	2
					Total MNPLR: 10,466.9 sccm
					Total MNPLR: 0.024 weight percent per day

SCCM – Standard Cubic Centimeters per Minute

The 95 percent upper confidence limit for the 2009 ILRT was 0.1195 weight percent per day. Combining the 95 percent upper confidence limit with the MNPLR for the Type B and C pathways provides the performance leakage rate. The performance leakage rate for the 2009 ILRT was 0.1435 weight percent per day, which is less than 1.0 L_a (0.20 weight percent per day). The ANSI/ANS 56.8-1994 standard requires the ILRT to be performed at a test pressure not less than 0.96 of the calculated peak design basis accident containment internal pressure (P_{ac}) nor greater than containment design pressure. For those plants with a P_{ac} of 25 psig or less, P_{ac} minus 1 psi shall be the minimum test pressure for the duration of the ILRT. Design pressure for the PNPP containment is 15 psig. Calculated peak LOCA pressure for PNPP is 6.40, but for conservatism is defined as 7.8 psig. The average test pressure during the 2009 ILRT was 8.4394 psig, which met the ILRT test pressure requirements of ANSI/ANS 56.8-1994.

The results of the last Type A ILRT for PNPP was less than the maximum allowable containment leakage rate of 0.20 weight percent per day. Since the test was successful, the PNPP ILRT remained on an extended frequency. The current ILRT frequency for PNPP is 10 years.

3.3 Plant Specific Confirmatory Analysis

A plant specific confirmatory analysis was performed to provide a risk assessment of permanently extending the currently allowed containment Type A ILRT to fifteen years. Attachment 2 contains the plant specific risk assessment conducted to support this proposed change.

3.3.1 Methodology

The risk assessment follows the guidelines from:

1. NEI 94-01, Revision 3-A.
2. The methodology used in Electric Power Research Institute (EPRI) Topical Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994.
3. The NEI document "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," dated October 2001.
4. The NRC regulatory guidance on the use of probabilistic risk analysis (PRA) as stated in Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," as applied to ILRT interval extensions.
5. Risk insights in support of a request for a change of the plant's licensing basis as outlined in Regulatory Guide 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."
6. 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 (Accession Number ML020920100).
7. The methodology used in EPRI Report No. 1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325," dated October 2008.

Revisions to 10 CFR 50, Appendix J, Option B allow individual plants to extend the ILRT Type A surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests performed at least 24 months apart in which the calculated performance leakage rate was less than the limiting containment leakage rate of $1L_a$.

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, Revision 0, states that NUREG-1493 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 EPRI TR-104285.

The NRC report on performance-based leak testing, NUREG-1493, 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 for a representative PWR plant (that is, Surry), that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for PNPP.

The guidance provided in Appendix H of EPRI Report No. 1009325, Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," for performing risk impact assessments in support of ILRT extensions builds on the EPRI risk assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT change.

Containment leak-tight integrity is verified through periodic in-service inspections conducted in accordance with the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service 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.

NRC regulation 10 CFR 50.55a(b)(2)(ix)(E) requires licensees to conduct visual inspections of the accessible areas of the interior of the containment. The proposed amendment to follow the guidance of NEI 94-01 will require that visual examinations be conducted during the outage that the ILRT is being conducted and during at least three other refueling outages. These requirements will not be changed because of the extended ILRT 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 are also not affected by the change to the Type A test frequency.

In the NRC safety evaluation for NEI 94-01, Revision 2, the NRC concluded that the methodology in EPRI Report No. 1009325, Revision 2, was 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. Table 3.3.1-1 addresses each of the four limitations and conditions for the use of EPRI Report No. 1009325, Revision 2.

Table 3.3.1-1, EPRI Report No. 1009325 Revision 2 Limitations and Conditions	
Limitation/Condition (From Section 4.2 of SE)	Response
1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of Regulatory Guide 1.200 relevant to the ILRT extension application.	PNPP PRA technical adequacy is addressed in Section 3.3.2 of this license amendment request.
2.a 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.	Regulatory Guide 1.174, Revision 3, 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 core damage frequency (CDF) less than 1 x 10^{-6} per year and increases in large early release frequency (LERF) less than 1 x 10^{-7} per year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval from three in ten years to once in 15 years is estimated as 2.64 x 10^{-8} per year using the EPRI guidance. As such, the estimated change in LERF is determined to be very small using the acceptance guidelines of Regulatory Guide 1.174.

Table 3.3.1-1, EPRI Report No. 1009325 Revision 2 Limitations and Conditions

Limitation/Condition (From Section 4.2 of SE)	Response
2.b 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.	The effect resulting from changing the Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.0318 person-rem/year. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of less than or equal to 1.0 person-rem per year or less than or equal to 1 percent of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The results of this calculation meet these criteria. Moreover, the risk impact for the ILRT extension when compared to other severe accident risks is negligible.
2.c In addition, a small increase in CCFP [conditional containment failure probability] should be defined as a value marginally greater than that accepted in a previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point.	The increase in the conditional containment failure probability from the three in ten-year interval to one in 15-year interval is 0.875 percent. EPRI Report No. 1009325, Revision 2-A, states that increases in CCFP of less than or equal to 1.5 percent are very small. Therefore, this increase is judged to be very small.

Table 3.3.1-1, EPRI Report No. 1009325 Revision 2 Limitations and Conditions	
Limitation/Condition (From Section 4.2 of SE)	Response
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 L_a instead of 35 L_a.	The representative containment leakage for Class 3b sequences is 100 L_a based on the guidance provided in EPRI Report No. 1009325, Revision 2-A. This is more conservative than the earlier industry Type A test interval extension requests, which utilized 35 L_a for the Class 3B sequences.
4. A license amendment request (LAR) is required in instances where containment over-pressure is relied upon for ECCS performance.	The PNPP design conservatively assumes 0 psig containment pressure and maximum expected temperatures of the pumped fluids. Thus, no reliance is placed on pressure and/or temperature transients to assure adequate NPSH. Reference Section 3.1 of this Enclosure for details

3.3.2 Technical Adequacy of the Probabilistic Risk Analysis (PRA)

3.3.2.1 Internal Events PRA Quality Statement for Permanent 15-Year ILRT Extension

No open findings related to the internal events PRA model remain at PNPP.

The PNPP PRA underwent an independent assessment in accordance with the guidance of Appendix X (Reference 3) of NEI 05-04, NEI 07-12, and NEI 12-13 (References 4, 5, and 6) as well as the expectations of the NRC (Accession No. ML17121A271). The independent assessment also considered the lessons learned from the NRC staff observations (that is, review) of the three independent assessment-team pilot reviews (Accession No. ML17095A252).

The final report documents the review process conducted, the facts and observations reviewed for internal events, internal flooding, and LERF models and the seismic model, and the overall results. The independent assessment concluded that model upgrades

occurred, and resulted in a focused scope peer review of common cause modeling and human reliability analysis. No new findings resulted from that review, and ultimately the satisfaction (closure) of all outstanding findings identified in previous peer reviews associated with the PNPP internal events, internal flooding, and LERF models was achieved. Previous reviews included a 2008 gap analysis self-assessment (internal events) against the PRA Standard (2005 Revision), a 2011 focused large early release frequency peer review, and a 2012 focused internal flood PRA peer review. A focused scope peer review of the offsite power recovery modeling in 2015 did not result in any identified findings.

The PNPP independent assessment is documented as part of the PRA model process and is available for audit.

3.3.2.2 PRA Maintenance and Update

The PNPP PRA model of record and supporting documentation have been maintained as a living program, with updates directed every other refueling cycle (every four years) to reflect the as-built, as-operated plant. Interim updates may be prepared and issued in between regularly scheduled model updates on an as-needed basis. Typically, an interim revision would be used for an update that would cause a change in CDF of greater than 10 percent, a change in LERF of greater than 20 percent, or for changes that could significantly impact a risk informed application. Interim models are also released following focused scope peer reviews once the associated findings and suggestions have been addressed.

FENOC employs a multi-faceted, structured approach in establishing and maintaining the technical adequacy and plant fidelity of the PRA models for FENOC nuclear generation sites. This approach includes a procedure that documents the PRA maintenance and update process, as well as the use of self-assessments and independent peer reviews.

The FENOC risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plant. This process is defined in the FENOC PRA Program, which consists of a governing procedure titled, "Probabilistic Risk Assessment Program," and subordinate implementation procedures. The governing procedure serves as the higher tier procedure and establishes the FENOC PRA Program and provides administrative requirements for the maintenance and upgrade of the FENOC PRA models and risk-informed applications. The overall objective of the PRA Program is to provide technically adequate PRA models such that the guidance set forth in Regulatory Guide 1.200, Revision 2, is satisfied for use in risk-informed applications. PRA related documentation is maintained within the records capture process.

In addition to model control, administrative mechanisms are in place to assure that changes from plant modifications, procedure changes, changes to calculations, and industry operating experience are appropriately screened, dispositioned, and tracked for incorporation into the PRA model if that change would impact the model.

3.3.2.3 Consistency with Applicable PRA Standards

An internal gap assessment between ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," and ASME/ANS RA-Sb-2005, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," was performed in support of the Technical Specification Task Force (TSTF) standard technical specifications change traveler TSTF-425 license amendment request request-for-additional-information (RAI) responses (RAI 3, page 9 of Reference 7). Between ASME/ANS RA-Sb-2005 and ASME/ANS RA-Sa-2009, only minor changes were made in the internal events portion of the standard. No gaps were identified relative to the ASME/ANS RA-Sa-2009 requirements.

Based on satisfaction of the independent assessment process against the requirements of ASME/ANS RA-Sa-2009 and with closure of outstanding findings associated with the internal events model, in conjunction with no gaps being identified in the comparison of the standards used for previous peer reviews, the PRA internal events model is compliant with Regulatory Guide 1.200, Revision 2, for the scope of this application, and meets Capability Category II or above in the ASME PRA Standard (ASME/ANS RA-Sa-2009). As such, the PRA internal events model can support risk-informed applications requiring Capability Category I or II capabilities, including this application.

3.3.2.4 External Events PRA Quality Statement for Permanent 15-Year ILRT Extension

The PNPP PRA is a Level 1 and 2 model that includes internal events and internal floods. For external events such as fire, seismic, extreme winds and other external events, the risk assessments from the PNPP Individual Plant Examination of External Events (IPEEE) can be used for insights on changes to ILRT intervals.

External hazards were evaluated in the PNPP IPEEE submittal in response to the NRC IPEEE Program (Reference 8). The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose for the identification of potential plant vulnerabilities and the understanding of associated severe accident risks.

The results of the PNPP IPEEE study are documented in the PNPP Individual Plant Examination of External Events for Severe Accident Vulnerabilities Report submitted to the NRC by a letter dated July 22, 1996 (Reference 9). Each of the PNPP external event evaluations were reviewed by the NRC and compared to the requirements of

NUREG-1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” (Accession Number ML063550238). The NRC transmitted to FENOC on March 9, 2001 their Staff Evaluation Report of the PNPP IPEEE submittal (Reference 10).

3.3.2.4.1 Fire Analysis

The PNPP PRA does not include a fire model. Therefore, the results of the fire risk assessment performed for the IPEEE must be qualitatively assessed for impact of the ILRT extension on fire risk. The IPEEE fire risk analysis quantified a CDF impact by combining the frequency of fires and the probability of detection/suppression failure with the remaining safety function unavailabilities. A systematic approach was used to identify critical fire areas where fires could fail safety functions and pose an increased risk of core damage if other safety functions are unavailable. The IPEEE CDF due to fires is 3.1×10^{-5} per year, with the dominant risk being fires in the control room, switchgear rooms, and specific elevations of the Control Complex, Fuel Handling, and Turbine Building (Reference 9).

The sensitivity study, which incorporates the fire analysis, is discussed in detail in Section 5.3 of Attachment 2 to this submittal. The overall conclusions are discussed in Section 3.3.3 below.

3.3.2.4.2 Seismic Analysis

The PNPP IPEEE seismic risk analysis did not quantify a CDF impact, but performed a seismic margin assessment that showed that PNPP is capable of attaining shutdown conditions and maintaining these conditions for 72 hours following a review level earthquake (RLE) of 0.3g (Reference 9).

The seismic CDF values reported in Table D-1, “Seismic Core-Damage Frequencies Using 2008 USGS Seismic Hazard Curves,” of Generic Issue 199, “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment,” (Accession No. ML100270756), are used for estimating seismic CDF in this calculation.

The sensitivity study, which incorporates the seismic analysis, is discussed in detail in Section 5.3 of Attachment 2 to this submittal. The overall conclusions are discussed in Section 3.3.3 below.

3.3.2.4.3 High Winds, Floods, and Other External Hazards

The risk of other external events such as high winds, floods, aircraft accidents, hazardous materials and turbine missiles was assessed in the PNPP IPEEE. The

IPEEE assessments concluded that the impact on PNPP from these events leads to the conclusion that there are no significant events of concern (Reference 9).

3.3.3 Summary of Plant-Specific Risk Assessment Results

Based on the results of the plant specific evaluation of risk and sensitivity calculations presented in Sections 5.2 and 5.3, respectively, of Attachment 2 to this submittal, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test frequency to 15 years:

- Regulatory Guide 1.174, Revision 3, 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 less than 1.0×10^{-6} per year and increases in LERF less than 1.0×10^{-7} per year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years is estimated as 2.64×10^{-8} per year using the EPRI guidance; this value increases negligibly if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included. As such, the estimated change in LERF is determined to be very small using the acceptance guidelines of Regulatory Guide 1.174.

- The effect resulting from changing the Type A test frequency to once-per-fifteen-years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.0318 person-rem per year. EPRI Report No. 1009325, Revision 2-A, states that a very small population dose is defined as an increase of less than or equal to 1.0 person-rem per year or less than or equal to 1 percent of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The results of this calculation meet these criteria. The risk impact for the ILRT extension when compared to other severe accident risks is negligible.

- The increase in the conditional containment failure probability from the 3 in 10 year interval to 1 in 15 year interval is 0.875 percent. EPRI Report No. 1009325, Revision 2-A, states that increases in CCFP of less than or equal to 1.5 percent are very small. Therefore, this increase is judged to be very small.

- Multiple sensitivity studies were performed as described in Section 5.3 of Attachment 2 of this submittal. As shown in Section 5.3.1, when both the internal and external event contributions are combined, the total change in LERF of 4.03×10^{-7} meets the guidance for a small change in risk, as it exceeds 1.0×10^{-7} per year and remains less than 1.0×10^{-6} change in LERF and the total LERF is

2.52 x 10^{-6}. Other sensitivity studies show the baseline ILRT extension analysis (as performed in Section 5.2 of Attachment 2 of this submittal) is conservative.

Therefore, the risk of increasing the ILRT interval to once per 15 years is considered to be small since it represents a small change to the PNPP risk profile.

3.4 Non-Risk Based Assessment

3.4.1 Containment and Drywell Vessel (Interior and Exterior) Visual Examination

The containment and drywell vessel (interior and exterior) visual examinations are performed to assess the general condition of the primary containment and drywell to detect evidence of degradation that may affect structural integrity. The examinations are performed to satisfy PNPP TS Surveillance Requirements 3.6.1.1.1 and 3.6.5.1.2. The examinations also satisfy the ASME Code, Section XI, Subsection IWE, Table IWE-2500-1, Item E1.11, Accessible Surface Areas examination requirements for examination of Class MC (metal containment) surfaces.

The containment structure examination is performed prior to the Type A test and at least once during each inservice inspection period. Drywell examinations are performed 3 times in 10 years at equal intervals. The visual examination of the containment accessible surfaces is performed using the VT-3M examination technique. Qualifications for visual examination of the drywell are handled in accordance with American National Standards Institute, American Society for Nondestructive Testing (ANSI/ASNT) Standard CP-189-1995, "Standard for Qualification and Certification of Nondestructive Testing Personnel," as referenced by ASME Code, Section XI or ANSI N45.2.6.

The examinations are conducted to identify the following items: structural deformation or degradation such that the component's function is impaired; missing or detached items; inservice cracked or flaking concrete (normally found shrinkage cracks are not inservice); excessive erosion or corrosion of structural metal which exceeds 10 percent of the design wall thickness; gross flaking, peeling or blistering of coatings; loosened items; excessive debris accumulation; and visual indication of degradation observed during the examination.

<u>Inspections of Drywell Coatings</u>

The PNPP Design Engineering Section concluded in 2001 that the inspections of the drywell coatings were redundant and unnecessary. Changes or elimination of the inspections was based on the following:

- Coating performance continues to remain acceptable regarding its design basis function to prevent corrosion and to facilitate decontamination. Gross degradation of drywell protective coatings has not been observed. Reasonably, gross failure of the

drywell coating would still be identified and addressed under the PNPP corrective action program.

- The original limit of drywell degraded coatings was based on an allowable surface area of the previous strainer design to become blocked with coatings. The previous design strainer surface was 42 square feet. The previous strainer approach velocity was above 0.35 feet per second, well above the average velocity of most coating debris. The current designed strainer surface area is greater than 1000 square feet per ECCS system, and the approach velocity of less than 0.02 feet per second is well below the average settling velocities of coating debris.

- A test was performed for the PNPP strainer design that combined LOCA generated debris with coating debris. The test allowed for 100% failure of drywell coatings (a total of 30 tons of coating debris). The results indicated increased strainer head loss, but the total strainer head loss was well within NPSH margins available for the ECCS pumps.

- The drywell is still required to be cleaned and inspected for cleanliness prior to drywell close-out, removing any coating debris that may have fallen to the floor.

Therefore, formal coatings inspections by the PNPP Design Engineering Section have been discontinued. Coatings continue to be monitored under the inservice inspection and maintenance rule programs.

<u>Unqualified/Degraded Coatings in Containment</u>

PNPP has minimal unqualified coatings in the containment or drywell. Some coatings touch-up work was performed for which design basis accident qualification data were only available to 307°F. The postulated presence of these unqualified coatings in the suppression pool and ECCS suction strainer is not design-limiting. Recent strainer head loss testing has indicated that significantly greater failed coatings quantifies could be tolerated.

The PNPP containment and drywell are periodically inspected for coatings in need of repair. Degraded coatings are routinely documented and tracked in coatings logs under the FENOC Corrective Action Program. Periodic inspections over the years have shown that coatings are not progressively delaminating at PNPP. A maintenance repetitive task initiates coating repairs when the amount of degraded coatings reaches 10 square feet. Therefore, the amount of deficient coatings in the PNPP drywell is currently very low and is maintained as such. On this basis, it is concluded that the amount of degraded coatings in the drywell is currently a negligible fraction of the total coating area. Under the present programs, this fraction will remain negligible.

3.4.2 Maintenance Rule Structure Monitoring Program

The Structure Monitoring Program performs periodic walk-downs of the structures within the scope of 10 CFR 50.65 (Maintenance Rule). Critical attributes of all structures in the scope of the rule have performance criteria that are predictive in nature and provide an early warning of degradation. This is accomplished by a combination of visual walk-down inspections and specific trending. Unavailability time will not be determined for structures; therefore, actual availability is not a performance criterion. The inspections are conducted with the intent of identifying defects that could prevent the structure from meeting its intended function and performance criteria as defined in the Maintenance Rule.

Frequency of Periodic Structure Walk-down Inspections

The frequency of the periodic structure walk-downs was initially set at two years. The frequency of periodic walk-down inspections can be adjusted based on a comparison with the results of the initial baseline walk-downs. When periodic walk-down inspections show no significant change in condition from that noted during the initial baseline walk-down or previous inspections, a decrease in the frequency will be evaluated. Conversely, if conditions get worse or new conditions develop, the frequency of inspections may be increased. The results of the initial baseline walk-down inspections are kept on file for comparison. The frequency of periodic structure walkdowns is currently every other refueling outage, nominally every four years.

The acceptance criteria for these inspections are as follows:

Acceptable (Category Y)

Acceptable structures capable of performing their structural functions, including protection of safety-related systems or components. Acceptable structures are free of deficiencies or degradation which could lead to possible failure. Presence of minor defects which are determined to be acceptable and which would not lead to possible failure is acceptable. These structures are categorized in accordance with Maintenance Rule as (a)(2).

Acceptable with Deficiencies (Category W)

Structures which are acceptable with deficiencies are those that are capable of performing their structural functions, including the protection or support of safety-related systems or components, but have degradation or deficiencies that could deteriorate to an unacceptable condition, if not analyzed or corrected prior to the next scheduled examination. A condition report is required and depending on the outcome of the analysis, these structures will be categorized in accordance with the Maintenance Rule as (a)(2) or (a)(1).

Unacceptable (Category N)

A structure is unacceptable if either (1) degradation is to the extent that the structure may not meet its design basis or (2) the structure has degraded to the extent that, if the degradation were allowed to continue uncorrected until the next normally scheduled examination, the structure may not meet its design basis. An unacceptable structure is categorized in accordance with the Maintenance Rule program as (a)(1) until the degradation and its cause have been corrected.

Structures identified as (a)(1) require a condition report with goal setting and monitoring to be performed. Because of the robust design and construction methods employed in nuclear power plants it would take extreme environmental conditions or many years of neglect for this condition to occur for building structures.

3.4.3 Containment Inservice Inspection Program

The Inservice Examination Plan (ISEP) delineates the components subject to non-destructive testing during the third inspection interval at PNPP, which commenced on May 18, 2009 and expires on May 17, 2019. Components requiring examination are vessels, pumps, valves, piping systems and their supports. Of the components requiring exams, certain parts such as welds, bolting, core support structures and interior surfaces may also require examination.

PNPP began commercial operation on November 18, 1987. Therefore, PNPP's first inspection interval was initially scheduled to expire on November 17, 1997. In accordance with IWA-2400(c) of the 1983 Edition with Summer 1983 Addenda (the Edition and Addenda applicable at that time), PNPP's first interval was extended 12 months (six months because the fourth refueling outage exceeded a 6 month duration and an additional six months for scheduling purposes) and expired on November 17, 1998.

PNPP's second inspection interval commenced on November 18, 1998 and the applicable code of record was ASME Code, Section XI, 1989 Edition for all Subsections except IWE and IWL. For subsections IWE and IWL, the 1992 Edition with the 1992 Addenda applied. For ultrasonic examinations, revisions to 10 CFR 50.55a published on September 22, 1999 (Final Rule) required the implementation of the ASME Code, Section XI, Division 1, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," of the 1995 Edition with 1996 Addenda as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Appendix VIII requires qualification of the procedures, personnel, and equipment used to detect and size flaws in piping, bolting and the reactor pressure vessel.

PNPP's second inspection interval was extended six months in accordance with IWA-2430(d) and expired on May 17, 2009.

PNPP's third inspection interval commenced on May 18, 2009. In accordance with 10 CFR 50.55a(g)(4)(ii), the inservice examination of components and system pressure tests shall be conducted during the successive 120-month inspection interval in compliance with the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b)(2) twelve months prior to the start of the interval. The referenced code edition and addenda for the third interval is ASME Code, Section XI, 2001 Edition through the 2003 Addenda.

In addition to the rules of ASME Code, Section XI, 2001 Edition through the 2003 Addenda, the following limitations and modifications required by 10 CFR 50.55a(b)(2) applied.

a. 10 CFR 50.55a(b)(2)(viii)(E) – For Class CC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the Inservice Inspection (ISI) Summary Report as required by IWA-6000:
    1. A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
    2. An evaluation of each area, and the result of the evaluation, and;
    3. A description of necessary corrective actions.

b. 10 CFR 50.55a(b)(2)(viii)(F) – Personnel that examine the containment concrete surfaces must meet the qualification provisions in IWA-2300. The "owner-defined" personnel qualification provisions in IWL-2310(d) are not approved for use.

c. 10 CFR 50.55a(b)(2)(ix)(A) – For Class MC applications, the following applies to inaccessible areas:
    1. The applicant or licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas.
    2. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report as required by IWA-6000: An evaluation of each area, and the result of the evaluation, and;
        i. A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
        ii. An evaluation of each area, and the result of the evaluation, and;

iii. A description of necessary corrective actions.

d. 10 CFR 50.55a(b)(2)(ix)(B) – When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

e. 10 CFR 50.55a(b)(2)(ix)(F) – VT-1 and VT-3 examinations must be conducted in accordance with IWA-2200. Personnel conducting examinations in accordance with the VT-1 and VT-3 examination method shall be qualified in accordance with IWA-2300. The "owner-defined" personnel qualification provisions in IWE-2330(a) for personnel that conduct VT-1 and VT-3 examinations are not approved for use.

f. 10 CFR 50.55a(b)(2)(ix)(G) – The VT-3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of Table IWE-2500-1, and the VT-1 examination method must be used to conduct the examination in Item E4.11 of Table IWE-2500-1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE-2500-1 using the VT-3 examination method must be conducted once each interval. The "owner-defined" visual examination provisions in IWE-2310(a) are not approved for use for VT-1 and VT-3 examinations.

g. 10 CFR 50.55a(b)(2)(ix)(H) – Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE-2500-1 must be examined using the VT-3 examination method. Flaws or degradation identified during the performance of a VT-3 examination must be examined in accordance with the VT-1 examination method. The criteria in the material specification or IWB-3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT-3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT-3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

h. 10 CFR 50.55a(b)(2)(ix)(I) – The ultrasonic examination acceptance standard specified in IWE-3511.3 for Class MC pressure-retaining components must also be applied to metallic liners of Class CC pressure-retaining components.

i. 10 CFR 50.55a(b)(2)(xxii) – The use of the provision in IWA-2200, "Surface Examination," of Section XI, 2001 Edition through the 2003 Addenda that allow use of an ultrasonic examination method is prohibited.

j. 10 CFR 50.55a(b)(2)(xxiv) – The use of Appendix VIII and the supplements to Appendix VIII and Article I-3000 of Section XI of the ASME Code, 2002 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section is prohibited. PNPP's ISI Program will follow the ASME Code, Section XI, 2001 Edition, no addenda for Appendix VIIII (including supplements) UT examinations as modified by 10 CFR 50.55a(b)(2)(xiv, xv and xvi).

Table 3.4.3-1 provides the inspection period and interval dates for the PNPP Third Ten Year Interval.

Table 3.4.3-1 PNPP Third Interval Refueling Outage Schedule May 18, 2009 – May 17, 2019			
Inspection Period	Refueling Outage	From	To
1	1R13 1R14	4/18/2011 3/18/2013	6/7/2011 5/16/2013
2	1R15 1R16	3/9/2015 3/5/2017	4/24/2015 4/3//2017
3	1R17	3/9/2019[1]	4/8/2019[1]

Note 1: The specified refueling outage month and day are approximate.

IWE/IWL (Class MC and CC) Examinations

The following abbreviations identify the unique nondestructive examination methods required for the examination area:

VT-1C – Detailed visual examination of Concrete Containment (Class CC) surfaces performed to determine the magnitude and extent of deterioration and distress of containment surfaces initially detected by an IWL General Visual examination (VT-3C). The visual examinations shall be performed directly with adequate illumination, by personnel who have been qualified for visual examination in accordance with IWA-2300 (that is, are certified VT-1), with visual acuity sufficient to resolve 0.044-inch lower case characters without ascenders or descenders (for example, a, c, e, o).

VT-1M – Detailed visual examination of Metal Containment (Class MC) surfaces performed to determine the magnitude and extent of deterioration and distress of containment surfaces requiring augmented inspection in accordance with IWE-1241 or suspect containment surfaces initially detected by an IWE General Visual examination (VT-3M). The visual examinations shall be performed directly with adequate illumination, by personnel who have been qualified for visual examination in accordance with IWA-2300 (that is, are certified VT-1), have a Class MC endorsement per FENOC's

nondestructive examination qualification procedure, and with visual acuity sufficient to resolve 0.044-inch lower case characters without ascenders or descenders (for example, a, c, e, o).

VT-3C – General visual examinations of Concrete Containment (Class CC) concrete surfaces conducted to determine the general structural condition of containments. The examination shall be performed in sufficient detail to identify areas of concrete deterioration and distress, such as defined in ACI 201.1 and ACI 349.3R. The visual examinations shall be performed directly or remotely, with adequate illumination, by personnel with visual acuity sufficient to detect evidence of degradation and who have been qualified for visual examination in accordance with IWA-2300 (that is, are certified VT-1 and/or VT-3 examiners per FENOC's nondestructive examination qualification procedure).

VT-3M – General visual examination of Metal Containment (Class MC) surfaces, performed either directly or remotely, by line of sight from available viewing angles from floors, platforms, walkways, ladders, or other permanent vantage points, to assess the general condition of containment surfaces. The examinations shall be performed with illumination sufficient to detect evidence of degradation. The examinations shall be performed by personnel who are qualified for visual examination (that is, are certified VT-1 or VT-3 examiners, or qualified General Visual examiners).

Class MC examination categories are identifiable by an 'E' as the first assigned letter and are as follows:

Table 3.4.3-2 IWE (Class MC) Examinations	
Examination Item No.[1]	Area (Examination Method)
Examination Category E-A Containment Surfaces	
E1.11	Accessible Surface Areas (VT-3M)
E1.11BC	Containment Bolted Connections (VT-3)[2]
E1.12	Wetted Surface of Submerged Areas (VT-3)
Examination Category E-C Containment Surfaces Requiring Augmented Examinations	
E4.11	Visible Surfaces (VT-1M)
E4.12	Surface Area Grid (Volumetric)

Notes:

1. Only those ASME Code item numbers that are applicable at PNPP.
2. A VT-1 Exam is required for bolted connections when flaws or degradation are identified.

Class CC examination categories are identifiable by an 'L' as the first assigned letter and are as follows:

Table 3.4.3-3 IWL (Class CC) Examinations	
Examination Item No.[1]	Area (Examination Method)
Examination Category L-A Concrete Surfaces	
L1.11	All Accessible Areas (VT-3C)
L1.12	Suspect Areas (VT-1C)
Examination Category L-B Unbonded Post Tensioning System	
Not Applicable at PNPP	

Notes:

1. Only those ASME Code item numbers that are applicable at PNPP.

Exemptions

Per the provisions of Section XI, components may be exempt from specific examination requirements if they are within the exemption criteria. Listed below are the exemptions applicable to the ISEP.

1. The following components (or parts of components) are exempted from the examination requirements of IWE-2000:
   a. Vessels, parts and appurtenances that are outside the boundaries of the containment as defined in the design specifications;
   b. Embedded or inaccessible portions of containment vessels, parts, and appurtenances that met the requirements of the original construction code;
   c. Portions of containment vessels, parts, and appurtenances that become embedded in concrete or become inaccessible as a result of vessel repair or replacement activities if the conditions of IWE-1232 and IWE-5220 are met;
   d. Piping, pumps and valves that are part of the containment system, or which penetrate or are attached to the containment vessel. These components shall be

examined in accordance with the rules of IWB or IWC, as appropriate to the classification defined by the design specifications.

2. The following components (or parts of components) may be exempted from the examination requirements of IWL-2000:

   a. Tendon end anchorages that are inaccessible, subject to the requirements of IWL-2521.1;

   b. Portions of concrete surface that are covered by the liner foundation material, or backfill, or are otherwise obstructed by adjacent structures, components, parts or appurtenances.

Examination Selection Process:

For Examination Categories E-A, E-C and L-A, ASME Code, Section XI, delineates criteria, which are applied in selecting the areas to be examined.

1. E-A Containment Surfaces

   The criteria for selecting containment surfaces are:

   a. VT-3M examination of the accessible interior and exterior containment surfaces of Class MC containment, which include:

      - integral attachments and structures that are parts of reinforcing structure, such as stiffening rings, manhole frames, and reinforcement around openings;

      - surfaces of attachment welds between structural attachments and the pressure retaining boundary or reinforcing structure, except for nonstructural or temporary attachments as defined in NE-4435 and minor permanent attachments as defined in CC-4543.4 of ASME Code Section XI.

   b. Pressure retaining bolted connections within the IWE boundary at least once per interval as required by 10 CFR 50.55a(b)(2)(ix)(G).

   c. At least 80 percent of the pressure-retaining boundary excluding attachments, structural reinforcement, and areas made inaccessible during construction.

   d. Including 100 percent of accessible wetted surfaces of submerged areas that are necessary, at least once per interval.

2. E-C Containment Surfaces Requiring Augmented Examinations

   The criteria for selecting the areas to be examined are:

   a. Interior and exterior containment surface areas that are subject to accelerated corrosion with no or minimal corrosion allowance or areas where the absence or

repeated loss of protective coatings has resulted in substantial corrosion and pitting. Typical locations of such areas are those exposed to standing water, repeated wetting and drying, persistent leakage, and those with geometries that permit water accumulation, condensation, and microbiological attack. Such areas may include penetration sleeves, surfaces wetted during refueling, concrete-to-steel shell or liner interfaces, embedment zones, leak chase channels, drain areas, or sump liners.

b. Interior and exterior containment surface areas that are subject to excessive wear from abrasion or erosion that causes a loss of protective coatings, deformation, or material loss. Typical locations of such areas are those subject to substantial traffic, sliding pads or supports, pins or clevises, shear lugs, seismic restraints, surfaces exposed to water jets from testing operation or safety relief valve discharges, and areas that experience wear from frequent vibrations.

3. L-A Concrete Surfaces

   100 percent of the accessible concrete surface area in the annulus shall be VT-3C visual examined for evidence of conditions indicative of damage or degradation each interval.

Successive Examinations (Reference IWE-2420)

The sequence of component exams established during the first inspection interval shall be repeated during each successive inspection interval, to the extent practical.

When examination results require evaluation of flaws or areas of degradation in accordance with IWE-3000, and the component is evaluated as acceptable for continued service, the areas containing such flaws or areas of degradation shall have augmented examination under Category E-C and re-examined during the next inspection period listed in the ISEP.

When augmented examinations under Category E-C reveal flaws or areas of degradation, which remain essentially unchanged for the next inspection period, these areas no longer require augmented examination under Category E-C and revert to the original schedule of successive exams.

Relief Requests

When compliance with ASME Code examination requirements is not practical, relief from examinations is requested. There are currently no Inservice Relief Requests (IR) required for the third 10-year inservice inspection interval for Subsections IWE and IWL.

Inservice Examination Table

The following tables contain a listing of Class MC and CC areas subject to the examination requirements of ASME Code, Section XI, Subsection IWE and IWL. This listing is for reference purposes only. The actual components scheduled for examination are presented to management for approval at least 60 days prior to commencing a scheduled refueling outage.

Abbreviations used for component descriptions in the tables are listed below.

AZ	- Azimuth
CTMT	- Containment
EL	- Elevation
FW	- Feedwater
INT	- Interior
No.	- Number
NS	- Not Scheduled
PEN	- Penetration
RM	- Room
&	- and
%	- Percent

Table 3.4.3-4 Inservice Examination Third Interval Listing Examination Category E-A

Item No.	Mark No.	Component Description	Exam Method	Period Schedule
E1.11	1T23-EXTERIOR			
E1.11	1T23-001-E	EXTERIOR EL 599-610 AZ 0-90 (2%)	VT-3M	1,2,3
E1.11	1T23-002-E	EXTERIOR EL 599-610 AZ 90-180 (2%)	VT-3M	1,2,3
E1.11	1T23-003-E	EXTERIOR EL 599-610 AZ 180-270 (2%)	VT-3M	1,2,3
E1.11	1T23-004-E	EXTERIOR EL 599-610 AZ 270-360 (2%)	VT-3M	1,2,3
E1.11	1T23-005-E	EXTERIOR EL 610-664 AZ 0-90 (9%)	VT-3M	1,2,3
E1.11	1T23-006-E	EXTERIOR EL 610-664 AZ 90-180 (7%)	VT-3M	1,2,3
E1.11	1T23-007-E	EXTERIOR EL 610-664 AZ 180-270 (9%)	VT-3M	1,2,3
E1.11	1T23-008-E	EXTERIOR EL 610-664 AZ 270-360 (7%)	VT-3M	1,2,3
E1.11	1T23-009-E	EXTERIOR EL 664-727 AZ 0-90 (9%)	VT-3M	1,2,3
E1.11	1T23-010-E	EXTERIOR EL 664-727 AZ 90-180 (9%)	VT-3M	1,2,3
E1.11	1T23-011-E	EXTERIOR EL 664-727 AZ 180-270 (9%)	VT-3M	1,2,3
E1.11	1T23-012-E	EXTERIOR EL 664-727 AZ 270-360 (9%)	VT-3M	1,2,3
E1.11	1T23-013-E	EXTERIOR DOME EL 727-757 (20%)	VT-3M	1,2,3
E1.11	1T23-014-E	EXTERIOR ANNULUS STEAM TUNNEL EL 616-649 (2%)	VT-3M	1,2,3
E1.11	1T23-015-E	EXTERIOR FUEL TRANSFER TUBE EL 620-652 (1%)	VT-3M	1,2,3
E1.11BC	1P53-A306-B	CTMT EQUIPMENT HATCH BOLTING	VT-3*	2

Table 3.4.3-4 Inservice Examination Third Interval Listing Examination Category E-A

Item No.	Mark No.	Component Description	Exam Method	Period Schedule
E1.11	1T23-INTERIOR	INTERIOR EL		
E1.11	1T23-005-I	INTERIOR EL 599-620 AZ 20-90 (2%)	VT-3M	1,2,3
E1.11	1T23-006-I	INTERIOR EL 599-620 AZ 90-180 (3%)	VT-3M	1,2,3
E1.11	1T23-007-I	INTERIOR EL 599-620 AZ 180-270 (3%)	VT-3M	1,2,3
E1.11	1T23-008-I	INT EXCEPT STEAM TUNNEL EL 599-620 AZ 270-340 (2%)	VT-3M	1,2,3
E1.11	1T23-009-I	INT EXCEPT FUEL TRANSFER EL 620-642 AZ 20-90 (2%)	VT-3M	1,2,3
E1.11	1T23-010-I	INT EXCEPT FUEL STORAGE PIT EL 620-642 AZ 90-180 (2%)	VT-3M	1,2,3
E1.11	1T23-011-I	INT EXCEPT FUEL TRANSFER EL 620-642 AZ 180-270 (3%)	VT-3M	1,2,3
E1.11	1T23-012-I	INTERIOR EL 620-642 AZ 270-340 (2%)	VT-3M	1,2,3
E1.11	1T23-013-I	INT EXCEPT STEAM TUNNEL & FILTER RM EL 642-690 AZ 0-90 (5%)	VT-3M	1,2,3
E1.11	1T23-014-I	INT EXCEPT FUEL STORAGE PIT EL 642-690 AZ 90-180 (4%)	VT-3M	1,2,3
E1.11	1T23-015-I	INTERIOR EL 642-690 AZ 180-270 (6%)	VT-3M	1,2,3
E1.11	1T23-016-I	INT EXCEPT STEAM TUNNEL EL 642-690 AZ 270-360 (6%)	VT-3M	1,2,3

Table 3.4.3-4 Inservice Examination Third Interval Listing Examination Category E-A

Item No.	Mark No.	Component Description	Exam Method	Period Schedule
E1.11	1T23-017-I	INT ABOVE POLAR CRANE EL 690-727 AZ 0-90 (4%)	VT-3M	1,2,3
E1.11	1T23-018-I	INT ABOVE POLAR CRANE EL 690-727 AZ 90-180 (4%)	VT-3M	1,2,3
E1.11	1T23-019-I	INT ABOVE POLAR CRANE EL 690-727 AZ 180-270 (4%)	VT-3M	1,2,3
E1.11	1T23-020-I	INT ABOVE POLAR CRANE EL 690-727 AZ 270-360 (4%)	VT-3M	1,2,3
E1.11	1T23-021-I	INTERIOR OF DOME EL 727-757 0-360 (19%)	VT-3M	1,2,3
E1.11	1T23-022-I	CONTAINMENT STEAM TUNNEL EL 616-649 AZ 340-20 (2%)	VT-3M	1,2,3
E1.11	1T23-023-I	FUEL TRANSFER TUBE EL 631-641 AZ 155-182 (<1%)	VT-3M	1,2,3
E1.11FH	1T23-FW-P121-FH	FW CONTAINMENT PEN. 121 FLUED HEAD SURFACES	VT-3M	NS+
E1.11FH	1T23-FW-P414-FH	FW CONTAINMENT PEN. 414 FLUED HEAD SURFACES	VT-3M	NS+
E1.11FH	1T23-MS-P122-FH	MAIN STEAM CONTAINMENT PEN. 122 FLUED HEAD SURFACES	VT-3M	NS+
E1.11FH	1T23-MS-P124-FH	MAIN STEAM CONTAINMENT PEN. 124 FLUED HEAD SURFACES	VT-3M	NS+
E1.11FH	1T23-MS-P415-FH	MAIN STEAM CONTAINMENT PEN. 415 FLUED HEAD SURFACES	VT-3M	NS+
E1.11FH	1T23-MS-P416-FH	MAIN STEAM CONTAINMENT PEN. 416 FLUED HEAD SURFACES	VT-3M	NS+
E1.11FH	1T23-MSD-P423-FH	MAIN STEAM DRAIN CONTAINMENT PEN. 423 FLUED HEAD SURFACES	VT-3M	NS+

Table 3.4.3-4 Inservice Examination Third Interval Listing Examination Category E-A

Item No.	Mark No.	Component Description	Exam Method	Period Schedule
E1.11FH	1T23-RCIC-P123-FH	REACTOR CORE ISOLATION COOLING CONTAINMENT PEN. 123 FLUED HEAD SURFACES	VT-3M	1,2,3
E1.11FH	1T23-RCIC-P422-FH	REACTOR CORE ISOLATION COOLING - RESIDUAL HEAT REMOVAL CONTAINMENT PEN. 422 FLUED HEAD SURFACES	VT-3M	NS+
E1.11FH	1T23-RHR-P421-FH	RESIDUAL HEAT REMOVAL CONTAINMENT PEN. 421 FLUED HEAD SURFACES	VT-3M	NS+
E1.11FH	1T23-RWCU-P131-FH	REACTOR WATER CLEAN-UP CONTAINMENT PEN. 121 FLUED HEAD SURFACES	VT-3M	NS+
E1.12	1T23-WETTED SURFACES OF SUBMERGED AREAS			
E1.12	1T23-001-I	SUPPRESSION POOL WALL EL 575-599 AZ 0-90	VT-3	3
E1.12	1T23-002-I	SUPPRESSION POOL WALL EL 575-599 AZ 90-180	VT-3	3
E1.12	1T23-003-I	SUPPRESSION POOL WALL EL 575-599 AZ 180-270	VT-3	3
E1.12	1T23-004-I	SUPPRESSION POOL WALL EL 575-599 AZ 270-360	VT-3	3
E1.20	NO BWR VENT SYSTEMS WITHIN PNPP'S IWE BOUNDARY			

* VT-1 examination shall be performed for bolted connections when flaws or areas of degradation are identified.

\+ Not scheduled consistent with Interpretation Record Number XI-1-13-25.

General Note: IWE/IWL examinations may be performed along with TS containment visual examinations covered by ISI Instruction T23-T2400-5.

No containment surfaces require augmented examinations in accordance with ASME Code, Section XI, Table IWE 2500-1, Examination Category E-C, Item Nos. E4.11 and E4.12.

Table 3.4.3-5 Inservice Examination Third Interval Listing Examination Category L-A

Item No.	Mark No.	Component Description	Exam Method	Period Schedule
L1.11	1T23-016-E	ANNULUS CONCRETE SURFACE AZ 0-360	VT-3C	R13, 2*
L1.12	NO AREAS IDENTIFIED			

* In accordance with IWL-2410, the frequency of examination for concrete surfaces after 10 years of service is once every five years plus or minus one year. Examinations were performed during the thirteenth refueling outage (1R13, 2011), and sixteenth refueling outage (1R16, 2017).

3.4.4 Revised Inspection Requirements

With the implementation of the proposed change, TS 5.5.12 will be revised by replacing the reference to Regulatory Guide 1.163 with a reference to NEI 94-01, Revision 3-A. This will require a visual examination prior to initiating a Type A test, of accessible interior and exterior surfaces of the containment system for structural problems that may affect the containment structure leakage integrity or the performance of the Type A test. This inspection must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years in accordance with the following sections of NEI 94-01, Revision 3-A:

- Section 9.2.1, "Pretest Inspection and Test Methodology"
- Section 9.2.3.2, "Supplemental Inspection Requirements"

In accordance with the inspection requirements outlined in Section 9.2.1 and 9.2.3.2 of NEI 94-01 Revision 3-A, FENOC will revise the ASME Code, Section XI, IWE and IWL programs and the implementing procedure, "Containment and Drywell Vessel (Interior and Exterior) Visual" to meet the specified intervals of containment inspection at PNPP.

3.4.5 Primary Containment Leakage Rate Testing Program - Type B and Type C Testing Program

PNPP Types B and C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and containment isolation valves in accordance with 10 CFR 50, Appendix J, Option B and Regulatory Guide 1.163. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Types B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. In accordance with TS 5.5.12, the allowable maximum pathway total Types B and C leakage is less than 0.6 L_a (51,700 standard cubic centimeters per minute [sccm]) where L_a equals 86,300 sccm.

As discussed in NUREG-1493, Type B and Type C tests can identify the vast majority of all potential containment leakage paths. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

A review of the Type B and Type C test results from 2009 through 2017 for PNPP has shown substantial margin between the actual as-found and as-left outage summations and the regulatory requirements as described below:

- The as-found minimum pathway leak rate average for PNPP shows an average of 27.94 percent of 0.6 L_a with a high of 42.60 percent of 0.6 L_a.
- The as-left maximum pathway leak rate average for PNPP shows an average of 30.27 percent of 0.6 L_a with a high of 43.46 percent of 0.6 L_a.

Table 3.4.5-1 provides a refueling outage (RFO) LLRT data trend summary.

Table 3.4.5-1 PNPP Type B and C LLRT Combined As-Found / As-Left Trend Summary					
RFO	2009	2011	2013	2015	2017
	RFO 12	RFO 13	RFO 14	RFO 15	RFO 16
AF Min Path[1] (SCCM)	22,023.20	16,005.00	14,522.70	10,186.30	9,493.00
Fraction of 0.6 L_a[2] (percent)	42.60	30.96	28.09	19.70	18.36
AL Max Path[3] (SCCM)	14,177.60	14,237.90	18,661.79	22,467.60	8,694.35
Fraction of 0.6 L_a (percent)	27.42	27.54	36.10	43.46	16.82
AL Min Path[4] (SCCM)	11,634.10	11,185.40	10,481.10	9,774.20	5,503.50
Fraction of 0.6 L_a (percent)	22.50	21.64	20.27	18.91	10.65

Notes:

1. AF Min Path is as-found minimum pathway leak rate.
2. L_a is the maximum allowable leakage rate at the peak containment pressure.
3. AL Max Path is as-left maximum pathway leak rate.
4. AL Min Path is as-left minimum pathway leak rate.

3.4.6 Type B and Type C Local Leak Rate Testing Program Implementation Review

Over the past two refueling outages at PNPP, no Type B or C tested component has exceeded its administrative limit while on an extended interval.

The percentage of the total number of PNPP Type B tested components that are on extended performance-based test intervals is 88 percent.

The percentage of the total number of PNPP Type C tested components that are on extended performance-based test intervals is 57 percent.

The PNPP extended frequency component percentage for Type B components in excess of 85 percent clearly supports that the maintenance and corrective action programs are aggressive in addressing component failures.

Several factors contribute to the lower percentage for Type C components on extended frequency. Several of the containment isolation valves are components which are part of the secondary containment bypass leakage, which are restricted by a low overall leakage limit (4,340 sccm at 7.8 psig). Additionally, certain penetrations and systems are included in the leakage outside of containment program, which requires testing every refueling outage. Finally, certain penetrations contain valves that are categorized as reclosing relief devices. These valves are replaced every refueling outage with pre-tested spares and the in-situ valve is removed for testing. The replacement of these valves requires a local leak rate test, even though the other containment isolation valves in the penetration qualify for testing on an Option B frequency.

3.5 Operating Experience

During the conduct of the various examinations and tests conducted in support of the containment-related programs previously mentioned, issues that do not meet established criteria or that provide indication of degradation, are identified, placed into the site's corrective action program, and corrective actions are planned and performed.

For the PNPP primary containment, the following site specific and industry events have been evaluated for impact on PNPP:

- Information Notice (IN) 1992-20, "Inadequate Local Leak Rate Testing"
- IN 2004-09, “Corrosion of Steel Containment and Containment Liner”
- IN 2010-12, "Containment Liner Corrosion"
- IN 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner"
- Regulatory Issue Summary (RIS) 2016-07, “Containment Shell or Liner Moisture Barrier Inspection”

Each of these areas is discussed in detail in Sections 3.5.1 through 3.5.5, respectively.

3.5.1 Information Notice (IN) 1992-20, "Inadequate Local Leak Rate Testing"

The NRC issued IN 1992-20 to alert licensees of problems with local leak rate testing two-ply stainless steel bellows used on piping penetrations at four different plants: Quad Cities, Dresden Nuclear Station, Perry Nuclear Power Plant, and the Clinton Station. Specifically, LLRTs 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 common issue in the four events was the failure to adequately perform local leak rate testing on different penetration configurations leading to problems that were discovered during ILRT tests in the first three cases.

In the event at Quad Cities the two-ply bellows design was not properly subjected to LLRT pressure and the conclusion of the utility was that the two-ply bellows design could not be Type B LLRT tested as configured.

In the events at both Dresden and Perry, flanges were not considered a leakage path when the Type C LLRT test was designed. This omission led to a leakage path that was not discovered until the plant performed an ILRT test.

In the event at Clinton relief valve discharge lines that were assumed to terminate below the suppression pool minimum drawdown level were discovered to terminate at a level above that datum. These lines needed to be reconfigured, and the valves should have been Type C LLRT tested.

Discussion

On July 6, 1989, Type A testing at PNPP was suspended due to excessive leakage from a flange on the residual heat removal (RHR) system. On August 23, 1989, during review of the Type A test results, it was discovered that the Type B testing requirements for this flange and its counterpart on the other train had not been met since March 31, 1986. The failure to perform Type B testing of the RHR flanges was a deficiency in the LLRT program.

On March 3, 1992, the NRC issued IN 1992-20. The response to the IN for PNPP focused on the Quad Cities event where it was determined that a valid Type B test could not be performed on the two-ply bellows design installed as containment penetration bellows. Type B LLRTs performed on bellows assemblies at PNPP were reviewed in detail with no significant findings. LLRTs conducted during three refueling outages detected minimal leakage. A detailed review of the PNPP bellows assemblies was performed. The review found that PNPP does not have two-ply bellows.

On February 23, 1993, following the response to IN 1992-20 for PNPP, it was discovered during an LLRT program review that two RHR system flanges were tested with water even though these components are above the suppression pool minimum drawdown, and would be exposed to air in an accident scenario. This review was prompted by NRC concerns regarding containment integrity. Immediate corrective action included ultrasonic testing that showed no leakage from the flanges due to the presence of water.

On November 24, 1993, Licensee Event Report (LER) 93-017-00 was submitted to the NRC concerning failure to perform local leak rate testing on the RHR System test return lines. The event was attributable to a failure to consider all possible leakage paths when the leak rate test program was established. Contributing to the event were inadequate reviews of previous PNPP and industry events. A modification to the flanged orifices at PNPP installed a testable spacer that allowed testing of the flange for leak tightness. Additionally, test instructions were developed to perform testing of the two RHR system flanges discussed in the preceding paragraph.

In May 1993, PNPP engineering re-evaluated IN 1992-20 and identified that there was one containment penetration bellows constructed with a two-ply design. The review was prompted by a submittal from another utility regarding testing of their inclined fuel transfer system (IFTS) penetration bellows. PNPP engineering and NRC personnel met on May 28, 1993 to discuss the incident. The meeting concluded with an acceptance of previous leakage measured at PNPP IFTS bellows based on low measured leakage rates. A VT-3 visual examination of the bellows outer ply was performed to confirm that there are no visual defects associated with the bellows assembly. The results of the containment ILRT conformed the integrity of the bellows. An engineering modification was developed to provide an alternate means of testing the IFTS bellows to resolve the concern identified in the engineering evaluation.

3.5.2 IN 2004-09, "Corrosion of Steel Containment and Containment Liner"

The NRC issued IN 2004-09 to alert addressees to recent occurrences of corrosion in freestanding metallic containments and in liner plates of reinforced and pre-stressed concrete containments. Any corrosion (metal thinning) of the liner plate or freestanding metallic containment could change the failure threshold of the containment under a challenging environmental or accident condition. Thinning changes the geometry of the containment shell or liner plate and may reduce the design margin of safety against postulated accident and environmental loads. Recent experience has shown that the integrity of the moisture barrier seal at the floor-to-liner or floor-to-containment junctions is important in avoiding conditions favorable to corrosion and thinning of the containment liner plate material. Inspections of containment at the floor level, as well as at higher elevations, have identified various degrees of corrosion and containment plate thinning.

Discussion

PNPP's containment is a free-standing metallic containment. However, unlike the free-standing containments discussed in the information notice, PNPP's containment does not have a moisture barrier seal at the floor-to-containment junction. Instead, PNPP has a 23 foot 6-inch concrete pour in the containment annulus with compressible material at the concrete pour to steel containment vessel interface.

PNPP has experienced moisture problems in the containment annulus, including water intrusion into the compressible material between the annulus pour (that is, concrete pour) and the steel containment. Water intrusion in 1999 was attributed to discharge from the main steam isolation valve (MSIV) packing leak-off line into the containment annulus. Visual inspections were performed to monitor the integrity of the annulus concrete and the containment vessel. In 2000 moisture buildup in the annulus was identified and was determined to be caused by an MSIV packing leak. The packing leak was repaired. A similar annulus condition was identified in 2002. Ultimately, a design change was implemented in 2002 that eliminated the MSIV packing leak-off lines, which prevented further moisture intrusion into the annulus from this source.

Water intrusion in 2002 was discovered as a result of the annulus floor drain sump pump not working properly. This problem caused standing water at the annulus pit below the lower personnel airlock barrel penetration. The annulus drain sump pump discharge check valve was not operating properly which caused backflow into the sump and the resultant standing water problem. As a result of the standing water, rust blisters were observed along the interface of the annulus floor (that is, above the height of the compressible material) and the containment vessel wall. Water was also seeping down the seams of the interface (compressible material) to below the concrete floor level. During the investigation of this condition, compressible material was removed at the annulus pit area below the lower airlock barrel to allow for inspections to occur. A VT-1 visual examination was performed on the containment wall where the compressible material was removed. No corrosion was evident below the floor level. Concurrently, ultrasonic thickness examinations were performed on the inside containment wall at the same location. The results did not indicate any corrosion or loss of wall thickness. The annulus drain sump pump discharge check valve was also repaired and the pump was properly pumping down the sump.

Although PNPP does not have a similar containment design as the plants mentioned in the IN, similar moisture intrusion problems have been experienced at the containment-to-annulus pour interface. However, these problems have been addressed through the corrective action program, and no further action specific to the IN was taken.

3.5.3 IN 2010-12, "Containment Liner Corrosion"

IN 2010-12 was issued to alert plant operators to three events that occurred where the steel liner of the containment building was corroded and degraded. At the Beaver Valley and Brunswick plants, material had been found in the concrete, which trapped moisture against the liner plate and corroded the steel. In one case, it was material intentionally placed in the building and in the other case, it was foreign material, which had inadvertently been left in the form when the wall was poured. But the result in both cases was that the material trapped moisture against the steel liner plate leading to corrosion. In the third case, Salem, an insulating material placed between the concrete floor and the steel liner plate absorbed moisture and led to corrosion of the liner plate.

Discussion

IN 2010-12 identifies that corrosion of the containment liner can be the result of contact with organic material. This is of greater concern when the corrosion exists between the liner and concrete as visual inspection is not available. At PNPP, a large accessible area exists between the containment liner and the concrete containment building; this is referred to as the containment annulus region. This allows for visual inspections of the containment liner which is performed and documented under the Maintenance Rule Program and ISI containment inspections. In 2011, the design drawings of the containment building, penetrations, hatches, airlocks and the annulus region were reviewed, including all change documents posted against them for wood scaffolding or felt wrapping. No evidence was found of any such material in the design drawings. The original construction specifications for the affected structures and components were also reviewed and only one reference to the use of any organic materials was noted. The specification for the annulus concrete installation states that, "The metallic Containment Shell shall be used as formwork for concrete placed...". No wooden formwork should have been used between the containment liner and the concrete which was placed around it; this prevents the existence of intentionally placed wood between the concrete and the containment.

Materials which were found on the design documents include the following:

- Slippit Form Debonding Agent – nonorganic liquid used to debond concrete.
- Lubrite Pads – Metal pads used to reduce friction.
- Compressible Material – Closed cell neoprene used in various locations.
- Bisco Seals – Nonorganic material silicone material used in penetrations.
- Sheet Metal – Nonorganic material with various uses.
- Pecora GC-5 Synthacalk – Synthetic rubber sealant.

These materials do not adversely affect the concrete or steel structures.

A review of the existing condition reports and work order history was performed in 2011, and no evidence of organic material was noted. An issue with water collecting on the concrete floor in the annulus region and seepage into the compressible material against the containment vessel was previously documented and addressed in the corrective action program. These evaluations are also mentioned in the evaluation of NRC IN 2004-09. Past inspection reports of the containment liner show no indication of organic material in contact with the containment liner and no rust related to organic material. Inspection criteria include a visual examination for blistering, flaking paint, peeling, discoloration, signs of distress, rust, bulging. Inspections that identify areas where acceptance criteria are not met are documented in the corrective action program for disposition. Review of the design drawings and construction specifications did not identify use or installation of organic materials at the containment vessel.

There are existing design characteristics, program aspects and process specifications that prevent and detect unacceptable corrosion at PNPP. The existing design documents do not indicate the presence of material that can trap moisture against the containment liner plate and lead to corrosion. Current programs have inspections of the containment liner to identify containment corrosion.

3.5.4 IN 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner"

The NRC issued IN 2014-07 to inform the industry of issues concerning degradation of floor weld leak-chase channel systems of steel containment shell and concrete containment metallic liner that could affect leak-tightness and aging management of containment structures. Specifically, this IN provides examples of operating experience at some plants of water accumulation and corrosion degradation in the leak-chase channel system that has the potential to affect the leak-tight integrity of the containment shell or liner plate. In each of the examples, the plant had no provisions in its ISI plan to inspect any portion of the leak-chase channel system for evidence of moisture intrusion and degradation of the containment metallic shell or liner within it. Therefore, these cases involved the failure to perform required visual examinations of the containment shell or liner plate leak-chase systems in accordance with the ASME Code, Section XI, Subsection IWE, as required by 10 CFR 50.55a(g)(4).

The containment base-mat metallic shell and liner plate seam welds of pressurized water reactors are embedded in a 3 to 4 feet thick concrete floor during construction and are typically covered by a leak-chase channel system that incorporates pressurizing test connections. This system allows for pressure testing of the seam welds for leak-tightness during construction and also while in service, as required. A typical base-mat shell or liner weld leak-chase channel system consists of steel channel sections that are

fillet welded continuously over the entire bottom shell or liner seam welds and subdivided into zones, each zone with a test connection.

Each test connection consists of a small carbon or stainless-steel tube (less than 1-inch diameter) that penetrates through the back of the channel and is seal-welded to the channel steel. The tube extends up through the concrete floor slab to a small access (junction) box embedded in the floor slab. The steel tube, which may be encased in a pipe, projects up through the bottom of the access box with a threaded coupling connection welded to the top of the tube, allowing for pressurization of the leak-chase channel. After the initial tests, steel threaded plugs or caps are installed in the test tap to seal the leak-chase volume. Gasketed cover plates or countersunk plugs are attached to the top of the access box flush with the containment floor. In some cases, the leak-chase channels with plugged test connections may extend vertically along the cylindrical shell or liner to a certain height above the floor.

Discussion

This IN was reviewed for applicability and commonality at PNPP. PNPP, being a BWR/6 with a GE Mark III containment, does not have some of the systems or components described in the IN. The PNPP containment is a free-standing steel cylinder with a flat steel bottom on a concrete foundation and a steel dome. Additionally, the lower region between the inside containment wall and outside drywell wall contains a suppression pool.

PNPP does have a leak chase system. There is a system embedded in concrete for the upper containment pools, but it is not part of containment integrity. There is also a leak chase system welded to the bottom of containment; however, it is submerged in the suppression pool. This allows some visual examination to be performed during IWE examinations of the wetted surfaces (visual exams of suppression pool surfaces). Containment shell welds are inside of the leak chase channels, making them inaccessible. These channels are exposed to demineralized water and constructed of stainless steel, so corrosion will be minimal or nonexistent.

Due to the above, PNPP is not affected by IN 2014-07.

3.5.5 Regulatory Issue Summary (RIS) 2016-07, “Containment Shell or Liner Moisture Barrier Inspection”

The NRC staff identified several instances in which containment shell or liner moisture barrier materials were not properly inspected in accordance with ASME Code, Section XI, Table IWE-2500-1, Item E1.30. Note 4 (Note 3 in editions before 2013) for Item E1.30 under the “Parts Examined” column states, “Examination shall include moisture barrier materials intended to prevent intrusion of moisture against inaccessible areas of the pressure retaining metal containment shell or liner at concrete-to-metal

interfaces and at metal-to-metal interfaces which are not seal welded. Containment moisture barrier materials include caulking, flashing and other sealants used for this application."

Examples of inadequate inspections have included licensees not identifying sealant materials at metal-to-metal interfaces as moisture barriers because they do not specifically match Figure IWE-2500-1, and licensees not inspecting installed moisture barriers, as required by Item E1.30, because the material was not included in the original design or was not identified as a "moisture barrier" in design documents.

Discussion

A review was performed of the current Inservice Examination Plan (ISEP). The ISEP specifically states that there are no moisture barriers (ASME Code, Section XI, Item No. E1.30) at PNPP. A review of design documents and condition reports was performed. Specification SP-801, "Procurement, Placement, and Testing of Annulus Concrete," is the historical construction specification for the "annulus fix," the ASME Class CC concrete pour between the shield building and metal containment vessel. This specification makes reference to the use of a neoprene sponge material placed between concrete and metal vessel. This material is not referred to as a moisture barrier within this specification, and appears to function as a gap filler to allow the concrete to expand and contract. Additionally, Section 5.03 of SP-801, Item 2 specifically states that the supply or installation of caulk at Elevation 598 feet 4 inches adjacent to the containment vessel is not in the scope of work. A review of construction specification SP-660, "Design and Fabrication of Steel Containment Vessels and Related Items for Reactor Buildings 1 & 2," for the metal containment has no reference to moisture barriers of any sort.

A review of condition reports indicates that water intrusion in the annulus has seeped between concrete and metal containment, indicating that there is no moisture barrier present. No moisture barriers have been installed; since, as the sources of the water intrusion (annulus sumps and MSIV packing leak-off) have been corrected or removed; installation of such a barrier is unnecessary.

A walk-down was performed on July 25, 2016. No moisture barriers were detected. Only the neoprene sponge material was observed. As discussed in design specifications and previous condition reports, the material extends above the concrete several inches and appears to be 3 inches thick (three, 1-inch pieces).

Based on the above information, no moisture barriers exist at PNPP and the ISEP plan is accurate (no E1.30 items).

3.5.6 Results of Recent Containment Inspections

2012 Containment and Drywell (Interior and Exterior) Visual Examination

The following conditions were identified during the 2012 Containment (interior) and Drywell (exterior) Visual Examination. Specified area dimensions and locations are approximate values taken from related condition reports.

Containment Interior, Elevation 599 feet to 620 feet:

1. Two areas 1.5 feet long by 4 inches wide where the paint has peeled or flaked away were identified at Elevation 599 feet between 24 and 32 degrees. The areas have moderate surface rust but no evidence of loose paint.

2. Three 2 inch square areas where the paint has peeled or flaked away were identified at Elevation 601 feet and 30 degrees. The areas have light surface rust but no evidence of loose paint.

3. An area 1 foot long by 4 inches wide where the paint has peeled or flaked away was identified at Elevation 608 feet and 30 degrees. This area has moderate surface rust but no evidence of loose paint.

4. Seventeen pipe supports located approximately at Elevation 615 feet between 34 degrees and 89 degrees have approximately 1 square foot baseplates. Running clockwise from 34 degrees, the second through sixth baseplates have moderate surface rust around their entire periphery that is about 2 inches wide. Continuing clockwise, the twelfth through seventeenth baseplates have light surface rust around their entire periphery that is about 1 inch wide. Paint on the attachment welds for the other baseplates is cracked in numerous places but is not flaking off.

5. An area 5 inches high by 16 inches wide where the paint has been ground away was identified at Elevation 607 feet and 76 degrees. This area has light surface rust but no evidence of loose paint around the edges.

6. An area 4 inches high by 6 inches wide where the paint has been chipped or flaked away was identified at Elevation 617 feet and 76 degrees. This area has light surface rust but no evidence of loose paint around the edges.

7. An area 4 inches by 4 inches where the paint has been chipped or flaked away was identified at Elevation 607 feet and 78 degrees. This area has light surface rust but no evidence of loose paint around the edges.

8. An area 5 inches high by 14 inches wide where the paint has been ground away was identified at Elevation 607 feet and 82 degrees. This area has light surface rust but no evidence of loose paint around the edges.

9. An area 4 inches by 4 inches where the paint has been chipped or flaked away was identified at Elevation 607 feet and 102 degrees. The area has light surface rust and no evidence of loose paint around the edges.

10. An area 2 inches high by 5 inches wide where the paint has been chipped or flaked away was identified at Elevation 607 feet and 108 degrees. The area has light surface rust and no evidence of loose paint around the edges.

11. An area 2 feet long by 2 inches wide where paint has been chipped or flaked away was identified at Elevation 607 feet and 130 degrees. The area appeared to be feathered out with no evidence of rust or loose paint around the edges.

12. An attachment weld for the Elevation 599 feet platform behind column CI-7 at approximately 144 degrees, has moderate surface rust for its visible length of approximately 2 feet. In most areas the weld is not visible as it is hidden by the platform but the platform is open behind the column. No signs of significant pitting or material loss were identified.

13. An area 2 feet long by 2 inches wide where the paint has been chipped or flaked away was identified at Elevation 608 feet and 165 degrees. The area appears to be feathered out and there is no surface rust or evidence of loose paint around the edges.

14. An area 6 inches wide by 5 inches high where the paint has been chipped or flaked away was identified at Elevation 608 feet and 168 degrees. The area has no rust and no evidence of loose paint around the edges.

15. An area 6 inches wide by 2 inches high where the paint has been chipped or flaked away and there is shiny metal was identified at Elevation 608 feet and 170 degrees. The area has no rust and no evidence of loose paint around the edges.

16. An area 5 inches wide by 4 inches high where the paint has been chipped or flaked away was identified at Elevation 607 feet and 178 degrees. The area has no rust, but there is some loose paint around the edges.

17. An area 3 inches high by 5 inches wide where the paint has been chipped or flaked away was identified at Elevation 601 feet and 188 degrees. The area has no rust or evidence of loose paint.

18. An area 2 feet long by 2 inches wide where the paint has been chipped or flaked away was identified at Elevation 601 feet and 190 degrees. The area appears to be feathered out and there is no surface rust or evidence of loose paint around the edges.

19. An area 20 inches long by 1 inch wide where the paint has been chipped or flaked away was identified at Elevation 601 feet and 190 degrees. The area appeared to be feathered out and there is no surface rust or evidence of loose paint around the edges.

20. An area 2 inches wide by 8 inches high on containment vertical seam 1-76, where the paint has been chipped or ground away, was identified at Elevation 601 feet and 222 degrees. The area has light surface rust and no evidence of loose paint around the edges.

21. An area 6 inches wide by 12 inches high where the paint has been ground away was identified at Elevation 615 feet and 224 degrees. This area has moderate surface rust, and no evidence of loose paint around the edges.

22. An area 6 inches wide by 12 inches high where the paint has been ground away was identified at Elevation 615 feet and 228 degrees. The area has moderate surface rust, and no evidence of loose paint around the edges.

23. An area 1.5 inches wide by 6 inches high where the paint has chipped or flaked away was identified at Elevation 608 feet and 228 degrees. This area has light surface rust, and no evidence of loose paint around the edges.

24. An area 6 inches wide by 8 inches high where the paint has chipped or flaked away was identified at Elevation 603 feet and 250 degrees. This area has no rust or evidence of loose paint.

25. An area 6 inches wide by 8 inches high in a past coatings repair where the paint has peeled or flaked away was identified at Elevation 603 feet and 256 degrees. This area has light surface rust, and no evidence of loose paint around the edges.

26. An area 4 inches wide by 6 inches high in a past coatings repair area where the paint is flaking and loose was identified at Elevation 602 feet and 258 degrees. This area has some evidence of light surface rust underneath.

27. Most of the paint has peeled or flaked away from a baseplate attachment weld and adjacent base material at Elevation 607 feet and 261 degrees. This area has light surface rust. Two other baseplates in the area have some spots where the paint has peeled or flaked away.

28. Two areas 6 inches apart from each other, 10 inches and 8 inches in diameter, were identified at Elevation 615 feet and 294 degrees. These areas have cracked and peeling paint.

29. An area 3 inches wide by 6 inches high where the paint has chipped or flaked away was identified at Elevation 603 feet and 298 degrees. This area has light surface rust, and no evidence of loose paint around the edges.

30. Paint has peeled or flaked away from the periphery of two support baseplates located at Elevations 603 feet and 606 feet and 318 degrees. These areas have light surface rust, and no evidence of loose paint.

31. An area 4 inches long by 2 inches wide where the paint is cracked and peeling along the bottom of a baseplate was identified at Elevation 613 feet and 330 degrees.

32. Three areas (4 inches long by 2 inches wide, 2 inches high by 4 inches wide, and 2 inches square) along two support baseplates at Elevation 601 feet and 334 degrees have paint that has peeled or flaked off. These areas have moderate surface rust. Paint is cracked along the periphery of the entire right side of the right side base plate.

33. An area 4 inches long by 2 inches wide along the periphery of a baseplate at Elevation 615 feet and 334 degrees has paint that is cracked and peeling.

34. An area 2 inches by 2 inches where the paint has flaked off was identified at Elevation 611 feet and 336 degrees. This area has light surface rust, and loose paint around the edges.

35. An area 6 inches wide by 5 inches high was identified at Elevation 607 feet and 360 degrees where the paint is cracked and peeling with evidence of moderate surface rust underneath. This indication is in the middle of an area where the coatings were previously repaired.

36. From 0 to 360 degrees there are approximately 200 areas less than one square inch in size where the paint has been chipped away by tooling or scaffolding impacts during refueling outages. Most of the areas have been feathered out and do not have any signs of rust. Some of the chipped areas only involve the topcoat.

There were no indications noted that would affect the structural integrity of the containment vessel. The rust was scraped away in a couple representative areas and there was either no noticeable material loss or where there was pitting, it was measured to be less than 1/16 inch.

Containment Interior, Elevation 620 feet to 690 feet:

1. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 633 feet and 149 degrees.

2. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 626 feet and 149 degrees.
3. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 624 feet and 149 degrees.
4. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 621 feet and 150 degrees.
5. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 625 feet and 151 degrees.
6. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 621 feet and 151 degrees.
7. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 628 feet and 152 degrees.
8. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 628 feet and 153 degrees.
9. Paint on an attachment weld is cracking and flaking with light surface rust on bare metal at Elevation 640 feet and 130 degrees.
10. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 641 feet and 135 degrees.
11. Paint is cracking and flaking in an area 6 inches by 12 inches at Elevation 642 feet and 142 degrees.
12. Paint is cracking and flaking in an area 5 inches by 10 inches at Elevation 638 feet and 143 degrees.
13. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 632 feet and 30 degrees.
14. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 632 feet and 33 degrees.
15. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 632 feet and 36 degrees.
16. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 632 feet and 40 degrees.
17. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 630 feet and 40 degrees.
18. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 632 feet and 42 degrees.

19. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 621 feet and 45 degrees.
20. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 621 feet and 50 degrees.
21. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 621 feet and 55 degrees.
22. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 630 feet and 60 degrees.
23. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 640 feet and 60 degrees.
24. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 634 feet and 60 degrees.
25. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 634 feet and 65 degrees.
26. Paint on an attachment weld is cracked in numerous places but is not yet flaking at Elevation 635 feet and 65 degrees.
27. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 636 feet and 66 degrees.
28. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 636 feet and 67 degrees.
29. Paint on an attachment weld is flaking with light surface rust on bare metal at Elevation 635 feet and 68 degrees.

The degradation of the coatings does not adversely impact the structural integrity of the containment. Most of the identified areas of minor rust are located on the attachment welds to containment and therefore do not adversely affect the containment structure.

Containment Interior, Elevation 690 feet to 727 feet

1. A 12 inch square support base plate and associated attachment welds were not coated at Elevation 713 feet and 35 degrees. The attachment welds and plate are covered with heavy rust. The 12 inch square plate is not welded directly to the containment vessel, rather, it is welded to a larger base plate.

2. Heavy surface rust was noted on base plate to vessel attachment support welds that were never coated at Elevation 724 feet and 37 degrees.

3. An area 2 inches by 3 inches where a paint blister had been blended out and not recoated leaving surface rust was identified at Elevation 726 feet and 95 degrees.

Just below this location there was a 1.5 inch circular spot where a paint blister existed and has been popped but not blended out. This location showed very light surface rust.

4. An area 1 inch round where the paint has been ground and there is surface rust on a shell circumferential weld was identified at Elevation 727 feet and 210 degrees.

5. An area 2 feet square with numerous spots of chipped paint from some sort of impact was identified at Elevation 723 feet and 240 degrees. Some of the chipped spots have minor surface rust, and others have no rust.

6. A 2-foot length of the attachment weld for a steel platform support where the surface has moderate rust was identified at Elevation 690 feet and 285 degrees.

7. An area 1 inch square where chipped or blistered paint has been buffed out was identified at Elevation 710 feet and 340 degrees. There is no rust on the bare metal.

8. Numerous small (typically, areas 0.75 inches square or less) chips in the paint from impacts made during refuel floor work activities over the years was identified at Elevation 690 feet to 696 feet and 360 degrees around. There was no evidence of loose paint, and the bare metal at the chipped locations was not rusted.

9. Paint on the containment vessel to dome circumferential weld is cracked, and there are signs of rust at Elevation 727.

10. An attachment weld at the top of a 16 inch by 16 inch unused baseplate (welded to the containment vessel) where the paint has cracked and there are signs of slight surface rust underneath was identified at Elevation 710 feet and 10 degrees.

11. An area 3 inches by 2 inches with a popped paint blister on a conduit support baseplate was identified at Elevation 696 feet and 90 degrees. The bare metal on the baseplate has light surface rust.

12. An area approximately 1 foot by 2 foot with several chipped paint spots was identified on the vessel wall at Elevation 691 feet and 92 degrees. The chipped paint spots have been feathered out, and there is light surface rust on the bare metal.

13. A 1 inch diameter paint blister with a star shaped crack was identified in the paint at Elevation 726 feet and 97 degrees.

14. A 1 inch diameter popped paint blister was identified at Elevation 722 feet and 140 degrees. The bare metal in the paint blister has no rust.

15. Chipped paint was identified in three spots at Elevation 702 feet and 175 degrees. All three spots are less than 1 inch square, and there was only very light surface rust on the bare metal.

16. A 1 inch diameter popped paint blister was identified at Elevation 725 feet and 184 degrees. The bare metal in the paint blister has no rust.

17. A small area where chipped paint has been blended out and there is a rust spot in the center was identified at Elevation 702 feet and 185 degrees. A similar indication was identified nearby at Elevation 694 feet and 195 degrees.

18. A 1 inch square area where the paint has been chipped or peeled away was identified at Elevation 706 feet and 235 degrees. There is no rust in this area.

19. A hangar support baseplate attachment weld with cracked paint and 2 square inches of paint that has peeled away was identified at Elevation 694 feet and 245 degrees. The area where the paint has peeled away has light surface rust underneath.

20. Paint on conduit support to baseplate welds is cracked with evidence of rust underneath for conduit runs at Elevations 697 feet and 710 feet. Indications are not on the containment vessel itself.

The degradation of the coatings does not adversely impact the structural integrity of the containment. The rusted areas do not exhibit significant material loss and therefore do not affect the structural integrity of the containment structure.

Containment Interior, Elevation 727 feet to 757 feet

Two areas (one 6 inches high by 9 inches wide and the other 5 inches high by 4 inches wide) where the paint has been ground away and there is light surface rust were identified at Elevation 728 feet and 135 degrees. The grind areas appear to be unchanged from previous inspections.

The degradation of the coatings does not adversely impact the structural integrity of the containment. The rusted areas do not exhibit significant material loss and therefore do not affect the structural integrity of the containment.

Drywell Exterior Indications:

1. A 3 inch square area where the paint was cracked and peeling was identified at Elevation 618 feet and 50 degrees.

2. An area where the paint was ground away around a support was identified at Elevation 603 feet and 132 degrees. This area has light surface rust.

3. A rusted area where the drywell hatch hoist rail is attached to the drywell wall was identified at Elevation 612 feet and 220 degrees.

4. Rust around a conduit support above the drywell equipment hatch was identified at Elevation 614 feet and 227 degrees.

5. Flaked and peeling paint at baseplates for conduit supports was identified at Elevations 604 feet and 612 feet and 250 degrees.

6. From 0 to 360 degrees around the drywell exterior, and primarily from Elevation 599 feet to 607 feet, numerous less than 1 square inch areas were identified where the paint has chipped away by tooling or scaffolding impacts during refueling outages. None of the areas exhibited any more than light surface rust.

The degradation of the coatings does not adversely impact the structural integrity of the containment. The rusted areas do not exhibit significant material loss and therefore do not affect the structural integrity of the containment structure.

2013 Drywell Interior Indications

Drywell Interior, Elevation 630 feet to Ceiling

1. Missing top coat in numerous areas was identified from 180 degrees to 270 degrees. Some areas with missing top coat are greater than 1 square foot. Most locations had the loose paint removed and edges feathered out with abrasives.

2. An area 2 inches square with cracked and peeling paint was identified along the bottom of a baseplate at Elevation 632 feet and 245 degrees.

3. Some areas with loose paint around their edges were identified above two large conduits at about 220 degrees. The areas were conservatively estimated to have 6 square inches of loose paint.

4. Small areas with cracked and peeling paint were identified near the baseplate for two small vertical conduits at Elevation 640 feet and 220 degrees. The areas with loose paint were conservatively estimated to total 6 square inches.

5. Small areas of flaked and peeling paint were identified at Elevation 645 feet and 105 degrees. The total amount of loose paint was conservatively estimated to be about 1 square foot.

6. An area 2 inches wide by 6 inches high where the paint is cracked and peeling was identified at Elevation 642 feet and 90 degrees.

7. An area 4 inches square where the paint is cracked and peeling was identified at Elevation 634 feet and 30 degrees.

8. Several areas of cracked and peeling paint were identified near the MSIV platform at 20 degrees. The total amount of loose paint is conservatively estimated to be about 1 square foot.

9. Numerous areas of loose paint near the MSIV platform were identified at Elevation 628 feet to 638 feet at 15 degrees. A number of these areas were seen in previous inspections but have not been repaired or feathered out. The total amount of loose paint was conservatively estimated to be about 3 square feet.

10. An area 2 inches square with cracked and peeling paint was identified at Elevation 634 feet and 315 degrees.

11. Two ceiling seam cover plates where the paint is flaked off and some areas with loose or peeling paint were identified on the drywell ceiling at about 0 degrees and 10 degrees. The total amount of loose paint for the two areas is conservatively estimated to be no more than 1/2 square foot.

12. Loose and peeling paint around the edges of a number of spots where paint is missing on the drywell ceiling were identified above a structural steel support at about 45 degrees. The total amount of loose paint is conservatively estimated to be 1 square foot.

13. Loose and peeling paint near a structural steel support was identified on the drywell ceiling at 60 degrees. The total amount of loose paint is conservatively estimated to be 1 square foot.

14. Loose and peeling paint was identified around an area near a structural steel support on the drywell ceiling at 60 degrees. The total amount of loose paint is conservatively estimated to be 1/2 square foot.

15. An area 4 inches wide by 10 inches long where paint is cracked and peeling was identified on the drywell ceiling near a structural steel support at 90 degrees. The total amount of loose paint was conservatively estimated to be 1/2 square foot.

16. An area 8 inches in diameter where paint is cracked and peeling was identified on the drywell ceiling at 100 degrees. The total amount of loose paint was conservatively estimated to be 1/2 square foot.

17. An area 2 inches wide by 6 inches long where paint is peeling was identified on the drywell ceiling at 130 degrees. The total amount of loose paint was conservatively estimated to be 1/4 square foot.

18. Areas at two locations along a ceiling seam backing bar or cover plate where the paint has flaked off, or there are signs of loose or peeling paint were identified at 132 degrees. The total amount of loose paint for these two areas was conservatively estimated to be no more than 1/2 square foot.

19. An area 4 feet long where the paint has flaked off and there are signs of loose or peeling paint was identified along a ceiling seam backing bar or cover plate at 200 degrees. The total amount of loose paint is conservatively estimated to be 1 square foot.

20. An area 6 inches high and 5 inches wide where paint is missing, and with cracked and peeling paint around the edges was identified on the drywell wall at 200 degrees. The total area of loose paint was conservatively estimated to be 1/4 square foot.

21. An area 8 inches wide by 5 inches high where paint is missing, and with cracked and peeling paint around the edges was identified on the drywell wall at 245 degrees. The total area of loose paint was conservatively estimated to be 1 square foot.

The degradation of coatings does not adversely impact the structural integrity of the drywell.

2017 Containment and Drywell (Interior and Exterior) Visual Examination

The following conditions were identified during the 2017 containment and drywell (interior and exterior) visual examination:

1. Moderate surface corrosion along welded areas with no significant material wastage was identified on the exterior fuel transfer tube penetration at Elevation 620-652. The observed corrosion was consistent with previous examination.

2. Light surface rust was identified in areas where structural components, that is, bolting and steel supports, interfaced with the containment vessel on the fuel transfer tube penetration at Elevation 631-641, Azimuth 155-182. One scrape was 0.75 inches by 2 inches with depth to bare metal. No loss of material was identified on the bare metal.

The 2017 containment and drywell visual examination results for the structures described above were satisfactory and there was no structural deformation or degradation such that the components function is impaired. Other than the indications identified above, no changes from the previous examination were noted.

2013 Reactor Building Maintenance Rule Structural Inspection

Reactor Building – Interior Containment

Corrosion on steel supports in areas of weld still exist. There are no signs that corrosion is causing degradation of function. Some areas of missing coating and surface corrosion exist on Elevation 689 feet. An approximately 1 square-foot area of coating is peeling near the 300 degree azimuth.

Reactor Building – Annulus and Interior Shield Building

No significant cracking or efflorescence was noted. Oxidation or rust of steel embedments and platforms appears to be consistent with previous walk-downs and has not worsened. Shield building dome steel beams are heavily corroded but do not appear to be degraded from previous walk-downs. The steel containment building did not show signs of further degradation from previous walk-downs.

Reactor Building – Drywell

Degraded coatings were identified by the 2012 and 2013 containment and drywell (interior and exterior) visual examinations.

Reactor Building Shield Building – Roof and Exterior

Roof: Hairline cracks in the roof parapet have not shown signs of growth since noted during the 2009 inspection. Roof booting around the seismic gap was pliable and appeared to be in good condition.

Exterior: Minor light cracking was noted at the construction joints. These areas do not indicate growth and are typical in this type of structure.

The 2013 reactor building maintenance rule structural Inspections found the structures to be acceptable and capable of performing their structural functions including protection of safety-related systems or components.

2017 Reactor Building Maintenance Rule Structural Inspection

Reactor Building – Interior Containment

An inspection was performed from elevation 599 feet 9 inches to the refuel floor at elevation 689 feet 6 inches.

Elevation 599 feet – Piping, conduit and supports above the suppression pool show signs of corrosion, especially at connection points where coating was not applied after field installation. No significant corrosion or loss of section was identified. Flaking paint was found in small areas of the coating on the containment vessel around the 599 foot

elevation near azimuth 270 to 330. This condition was identified during previous ASME Code, Section XI, inservice inspections.

Other elevations in general:

1. Some areas had minor surface rust on supports and structural steel. These areas did not result in a loss of section.

2. Some small areas where the top coat of paint has worn away but the primer remains present. Especially on the refuel floor checkered plate. These areas did not appear to be flaking and do not present a hazard for foreign material exclusion in the pools at this time.

Reactor Building – Drywell

No cracking or damage to the concrete was identified. Nothing beyond minor surface rust was identified on any structural steel. As identified during previous inspections, the nuclear closed cooling system remains rusted. However, this piping is outside of the scope of the Maintenance Rule Program. Loss of surface coating, but not primer, was identified in several areas throughout the drywell. Most locations were concentrated on the drywell liner. The findings were identical to those found during the 2009 inspections. The inspection did not include the under-vessel area due to radiological conditions.

Reactor Building – Annulus and Interior Shield Building

Indications observed were similar to previous inspections. Evidence of past water or moisture intrusion in the annulus was prevalent. Surface rusting was observed on structural steel platforms, grating, ladders, embed plates, and other items. Loss of material was not identified, consistent with previous inspections. No significant cracking or efflorescence was identified. Some unfilled bolt holes were identified which do not impact the structure.

Reactor Building Shield Building – Exterior and Roof (Dome)

Overall, no major issues were identified with the shield building roof, exterior concrete or steel (ladders). Typical hairline cracks in the parapet were observed. These cracks have been previously identified during the 2009 and 2013 inspections and have shown little to no growth. Such cracks are normal for concrete structures and do not present any concerns. In some areas, the copper grounding wire was detached from the parapet hooks.

Minor peeling of the coating on the roof dome was observed. Surface rusting was observed on the 38 inch containment vacuum relief vent and spare 14 inch penetration. Concrete spalling was observed occurring adjacent to the corner of the 14 inch

penetration. Metal has not been exposed but remains an item to monitor in future walk-downs.

Surface rusting was observed on the 38 inch spare penetration and concrete spalling was observed around portions of the penetration's perimeter. Hairline cracks and efflorescence were observed on the exterior wall above the intermediate building.

The 2017 reactor building maintenance rule structural inspections found the structures to be acceptable and capable of performing their structural functions including protection of safety-related systems or components.

3.6 Containment Modifications

Repair to Containment Equipment Hatch

During the Spring 2015 refueling outage, a crane mobilizing equipment contacted the PNPP containment equipment hatch. This resulted in damage to the face of the equipment hatch sealing surface. The defects were required to be removed and a weld build-up performed to restore the original dimensions of the sealing surface in order to ensure an adequate sealing surface and pressure boundary. The containment equipment hatch is a safety-related, Seismic Category I component. The containment vessel conforms to the requirements of ASME Code, Section III, Subsection NE for Class MC components. This activity was considered a repair and was required to meet the requirements of ASME Code, Section XI, 2001 edition with addenda through 2003.

The body ring of the containment equipment hatch is 3 inches thick and is constructed of ASME SA-516 Grade 70 steel. ASME Code, Section XI, Subarticle IWA-4400 requires welding to be performed in accordance with the original construction code. Required NDE after defect removal was dictated by ASME Code, Section XI, IWA-4611. Post repair inspections were dictated by ASME Code, Section XI, Subarticle IWE-2000.

In total, six defects were identified on the containment equipment hatch. Repair of the cavities began with grinding until the defects were smooth and clean with beveled sides and rounded edges to provide suitable accessibility for welding. The amount of material removed was minimized to that of the defect. Following grinding, surface (magnetic particle) examinations were performed on the defect removal cavity. The examinations revealed no extended cracks in the damaged area. Weld build up and machining was used to restore the configuration of the containment equipment hatch closure flange to the original configuration. A post weld VT-3M visual examination of the repair area was satisfactory.

Post maintenance testing was performed by a leak test procedure entitled: "Type B Local Leak Rate Test of 1P53 Containment Equipment Hatch, Penetration P202, and

the Penetration Pressurization System." The as-left leakage rate for the test of the hatch seals was 5.83 sccm. Acceptance criterion for the test was 250.0 sccm.

3.7 License Renewal Aging Management

By correspondence dated May 25, 2017 (Accession No. ML17145A171), FENOC revised the preparation and submittal schedule for the license renewal application for PNPP. As requested by the NRC in RIS 2009-06, "Importance of Giving NRC Advance Notice of Intent To Pursue License Renewal," FENOC notified the NRC of its intent to submit the PNPP license renewal application in the fourth quarter of 2020.

3.8 NRC Safety Evaluation Limitations and Conditions

3.8.1 Limitations and Conditions Applicable to Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 2-A

By letter dated June 25, 2008, the NRC staff issued a safety evaluation for NEI 94-01, Revision 2. The safety evaluation documents the NRC staff's evaluation and acceptance of NEI TR 94-01, Revision 2, subject to the limitations and conditions noted in Section 4.0 of the safety evaluation.

The limitations and conditions from the June 25, 2008 safety evaluation (SE) are presented below in Table 3.8.1-1 with the FENOC response for PNPP.

Table 3.8.1-1: NEI 94-01, Revision 2-A, Limitations and Conditions

Limitation/Condition (From Section 4.0 of SE)	Response
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, [Reference 11]. (Refer to SE Section 3.1.1.1.)	PNPP will utilize the definition in NEI 94-01 Revision 3-A, Section 5.0. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.
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.)	Reference Section 3.4.4 of this submittal.
The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3.)	Reference Section 3.4.3 of this submittal.

Table 3.8.1-1: NEI 94-01, Revision 2-A, Limitations and Conditions	
Limitation/Condition (From Section 4.0 of SE)	Response
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 major modifications planned. Past containment modifications are discussed in Section 3.6 of this submittal.
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.)	The requirements of NEI 94-01, Revision 3-A, Section 9.1 will be followed at PNPP. This requirement has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01. In accordance with Section 3.1.1.2 of the NRC safety evaluation dated June 25, 2008, FENOC will also demonstrate to the NRC staff that an unforeseen emergent condition exists in the event an extension beyond the 15-year interval is required at PNPP. Except for compelling reasons, which could include unforeseen conditions, the licensee will conduct the Type A tests within the approved 15-year interval without seeking extensions. If an unforeseen emergent condition should arise, extension of the Type A test interval will be addressed in accordance with NEI 94-01, Revision 2-A, the associated NRC safety evaluation, and RIS 2008-27 up to and including seeking NRC approval through a license amendment if required.
For plants licensed under 10 CFR 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	Not applicable. PNPP was not licensed under 10 CFR 52.

Table 3.8.1-1: NEI 94-01, Revision 2-A, Limitations and Conditions	
Limitation/Condition (From Section 4.0 of SE)	Response
the applicability of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	

3.8.2 Limitations and Conditions Applicable to Topical Report NEI 94-01, Revision 3-A

A safety evaluation dated June 8, 2012 documents the NRC staff's evaluation and acceptance of NEI TR 94-01, Revision 3, subject to the limitations and conditions summarized in Section 4.0 of the safety evaluation.

The limitations and conditions from the June 8, 2012 safety evaluation are presented below and followed by the FENOC response for PNPP.

Topical Report Condition 1

> NEI TR 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs be increased to 75 months with the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84-months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g., BWR MSIVs), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months.

Response to Condition 1

Condition 1 presents three separate issues that are required to be addressed. They are as follows:

- ISSUE 1 - The allowance of an extended interval for Type C LLRTs of 75 months carries the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit.

Response to Condition 1, ISSUE 1

The post-outage report shall include the margin between the Type B and Type C MNPLR summation value, as adjusted to include the estimate of applicable Type C leakage understatement, and its regulatory limit of 0.60 L_a.

- ISSUE 2 - A corrective action plan shall be developed to restore the margin to an acceptable level.

Response to Condition 1, ISSUE 2

When the potential leakage understatement adjusted Type B and C MNPLR total is greater than the PNPP leakage summation limit of 0.50 L_a, but less than the regulatory limit of 0.6 L_a, then an analysis and determination of a corrective action plan shall be prepared to restore the leakage summation margin to less than the PNPP leakage summation limit. The corrective action plan shall focus on those components that have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues to maintain an acceptable level of margin.

- ISSUE 3 - Use of the allowed 9-month extension for eligible Type C valves is only authorized for non-routine emergent conditions with exceptions as detailed in NEI 94-01, Revision 3-A, Section 10.1.

Response to Condition 1, ISSUE 3

The 9-month allowable interval extension period will only be applied to eligible Type C components and only for non-routine emergent conditions. Such occurrences will be documented in the record of tests.

Topical Report Condition 2

> The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time. The containment leakage condition monitoring regime involves a portion of the penetrations being tested each refueling outage, nearly all LLRTs being performed during plant outages. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as-found minimum pathway leakage rates for the just tested penetrations are summed with the as-left minimum pathway leakage rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Type C tests involve valves, which in the aggregate, will show increasing

leakage potential due to normal wear and tear, some predictable and some not so predictable. Routine and appropriate maintenance may extend this increasing leakage potential. Allowing for longer intervals between LLRTs means that more leakage rate test results from farther back in time are summed with fewer just tested penetrations and that total used to assess the current containment leakage potential. This leads to the possibility that the LLRT totals calculated understate the actual leakage potential of the penetrations. Given the required margin included with the performance criterion and the considerable extra margin most plants consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI TR 94-01, Revision 3, Section 12.1.

When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Response to Condition 2

Condition 2 presents two (2) separate issues that are addressed as follows:

- ISSUE 1 - Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI 94-01, Revision 3, Section 12.1.

Response to Condition 2, ISSUE 1

The change in going from a 60-month extended test interval for Type C tested components to a 75-month interval, as authorized under NEI 94-01, Revision 3-A, represents an increase of 25 percent in the LLRT periodicity. As such, a potential leakage understatement adjustment factor of 1.25 will be conservatively applied to the actual as-left leak rate, which will increase the as-left leakage total for each Type C component currently on greater than a 60-month test interval up to the 75-month extended test interval. This will result in a combined conservative Type C total for all 75-month LLRTs being carried forward and included whenever the total leakage summation is required to be updated (either while operating on-line or following an outage).

When the potential leakage understatement adjusted leak rate total for those Type C components being tested on greater than a 60-month test interval up to the 75-month extended test interval is summed with the non-adjusted total of those Type C components being tested at less than or equal to a 60-month test interval, and the total of the Type B tested components, if the MNPLR is greater than the PNPP leakage summation limit of 0.50 L_a, but less than the regulatory limit of 0.6 L_a, then an analysis and corrective action plan shall be prepared to restore the leakage summation value to less than the PNPP leakage limit. The corrective action plan shall focus on those components that have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

- ISSUE 2 - When routinely scheduling any LLRT valve interval beyond 60 months and up to 75 months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B and C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Response to Condition 2, ISSUE 2

If the potential leakage understatement adjusted leak rate MNPLR is less than the PNPP leakage summation limit of 0.50 L_a, then the acceptability of the greater than a 60-month test interval up to the 75-month LLRT extension for affected Type C components has been adequately demonstrated and the calculated local leak rate total represents the actual leakage potential of the penetrations.

In addition to Condition 1, Issues 1 and 2, which deal with the MNPLR Type B and C summation margin, NEI 94-01, Revision 3-A, also has a margin related requirement contained in Section 12.1, "Report Requirements."

> A post-outage report shall be prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage. The technical contents of the report are generally described in ANSI/ANS-56.8-2002 and shall be available on-site for NRC review. The report shall show that the applicable performance criteria are met, and serve as a record that continuing performance is acceptable. The report shall also include the combined Type B and Type C leakage summation, and the margin between the Type B and Type C leakage rate summation and its regulatory limit. Adverse trends in the Type B and Type C leakage rate summation shall be identified in the report and a corrective action plan developed to restore the margin to an acceptable level.

In the event an adverse trend in the potential leakage understatement adjusted Type B and C summation is identified, an analysis and determination of a corrective action plan shall be prepared to restore the trend and associated margin to an acceptable level. The corrective action plan shall focus on those components that have contributed the most to the adverse trend in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

An adverse trend is defined as three (3) consecutive increases in the final pre-mode change Type B and C MNPLR summation value adjusted to include the estimate of applicable Type C leakage understatement, as expressed in terms of L_a.

3.9 Conclusion

NEI 94-01, Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, describe an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporated the regulatory positions stated in Regulatory Guide 1.163 and includes provisions for extending Type A intervals to 15 years and Type C test intervals to 75 months. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. FENOC is proposing to adopt the guidance of NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, for the PNPP 10 CFR 50, Appendix J testing program plan.

The NEI 94-01, Revision 3-A, guidance includes provisions for extending Type C LLRT intervals up to 75 months. Extending Type C LLRT intervals in accordance with this guidance is acceptable based on performance history and risk insights.

Based on the previous ILRTs conducted at PNPP, it may be concluded that the permanent 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, and the overlapping inspection activities performed as part of the following PNPP inspection programs:

- Inservice Inspection Program IWE and IWL
- Containment Inspections per TS Surveillance Requirement 3.6.1.1.1
- Containment and Drywell Vessel (Interior and Exterior) Visual Examinations

The PNPP risk assessment provided in Attachment 2 concludes that the risk impact of increasing the ILRT interval to a 1 in 15-year frequency is considered to be small since it represents a small change to the PNPP risk profile.

4.0 REGULATORY EVALUATION

The proposed license amendment would revise Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program," to follow guidance developed by the Nuclear Energy Institute (NEI) and presented in topical report, NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," with conditions and limitations specified in NEI 94-01, Revision 2-A. The NEI 94-01 guidelines would allow an extension of the containment integrated leak rate test (Type A test) interval, and the containment isolation valve local leakage-rate test (Type C test) interval. The current Type A test interval of 120 months (10 years) would be extended on a permanent basis to no longer than 15 years from the last Type A test. The current Type C test interval of 60 months for selected components would be extended on a performance basis to no longer than 75 months. Extensions of up to nine months (total maximum interval of 84 months for Type C tests) would be permissible only for non-routine emergent conditions.

The proposed license amendment would also revise Technical Specification 5.5.12 by deleting two of the four listed exceptions to Primary Containment Leakage Rate Testing Program guidelines. The exceptions to be deleted are no longer necessary and involve the performance of a Type A test no later than a specified date, and corrections to NEI 94-01, Revision 0.

4.1 No Significant Hazards Consideration

FirstEnergy Nuclear Operating Company has evaluated whether 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 test interval extensions do not involve either a physical change to the plant or a change in the way the plant is operated or controlled. The containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment and the testing requirements invoked 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 the prevention or identification of

any precursors of an accident. Therefore, the proposed extensions do not involve a significant increase in the probability of an accident previously evaluated.

The effect resulting from changing the Type A test frequency to 1 per 15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.0318 person-rem/year. EPRI Report No. 1009325, Revision 2-A, states that a very small population dose is defined as an increase of less than or equal to 1.0 person-rem per year or less than or equal to 1 percent of the total population dose, whichever is less restrictive for the risk impact assessment of the extended integrated leak rate test intervals. The results of the risk assessment calculation for the Type A test extension meet these criteria. The risk impact for the integrated leak rate test extension when compared to other severe accident risks is negligible.

The integrity of the containment is subject to two types of failure mechanisms that can be categorized as: (1) activity based, and (2) time based. Activity based failure mechanisms are defined as degradation due to system and component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the containment combined with the containment inspections performed in accordance with ASME Code, Section XI, and Technical Specification requirements serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed test interval extensions do not significantly increase the consequences of an accident previously evaluated.

The proposed amendment also deletes two previously granted exceptions to Primary Containment Leakage Rate Testing Program guidelines. The exception regarding the performance of a Type A test no later than a specified date would be deleted as this Type A test has already been performed. Additionally, the exception to use the corrections to NEI 94-01, Revision 0, would be deleted as those corrections would no longer be in use. These changes to the exceptions in Technical Specification 5.5.12 are administrative in nature and do not affect the probability or consequences of an accident previously evaluated.

Therefore, the proposed changes do not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

Containment Type A and Type C testing requirements periodically demonstrate the integrity of the containment and exist to ensure the plant's ability to mitigate the consequences of an accident. These tests do not involve any accident precursors or initiators.

The proposed change does not involve a physical modification to the plant (that is, no new or different type of equipment will be installed) nor does it alter the design, configuration, or change the manner in which the plant is operated or controlled beyond the standard functional capabilities of the equipment.

The proposed amendment also deletes two previously granted exceptions. The exception regarding the performance of a Type A test no later than a specified date would be deleted as this Type A test has already been performed. Additionally, the exception to use the corrections to NEI 94-01, Revision 0, would be deleted as those corrections would no longer be in use. These changes to the exceptions in Technical Specification 5.5.12 are administrative in nature and do not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

   Response: No.

   The proposed license amendment does not alter the way safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the Technical Specification Primary Containment Leakage Rate Testing Program exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by Technical Specifications is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed amendment, since they are not affected by implementation of a performance-based containment testing program. This ensures that the margin of safety in the plant safety analysis is maintained.

   The proposed amendment also deletes two previously granted exceptions. The exception regarding the performance of a Type A test no later than a specified date would be deleted as this Type A test has already been performed. Additionally, the exception to use the corrections to NEI 94-01, Revision 0, would be deleted as those

corrections would no longer be in use. These changes to the exceptions in Technical Specification 5.5.12 are administrative in nature and do not involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, FirstEnergy Nuclear Operating Company concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4.2 Applicable Regulatory Requirements/Criteria

The proposed amendment has been evaluated to determine whether applicable regulations and requirements continue to be met.

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J includes two options, Option A, "Prescriptive Requirements," and Option B, "Performance-Based Requirements," either of which may be chosen for meeting the requirements.

The testing requirements in Appendix J to 10 CFR Part 50 ensure that: (a) leakage through the containments or systems, and components penetrating the containments, does not exceed allowable leakage rates specified in the Technical Specifications (TSs) and (b) the integrity of the containment structure is maintained during its service life.

Option B specifies the performance-based requirements and criteria for preoperational and subsequent leakage rate testing. These requirements are met by performance of Type A tests to measure the containment system overall integrated leakage rate, Type B pneumatic tests to detect and measure local leakage rates across pressure retaining leakage-limiting boundaries such as penetrations, and Type C pneumatic tests to measure containment isolation valve leakage rates. Preoperational tests are required to be conducted at periodic intervals to ensure integrity of the overall containment system as a barrier to fission product release. Type A tests intervals are based on the historical performance of the overall containment system, and Type B and Type C test intervals are based on historical performance of each boundary and isolation valve. The leakage rate test results must not exceed the maximum allowable leakage rate (L_a) with margin, as specified in the TSs.

On September 9, 1997, the NRC approved TS Amendment 86 for PNPP authorizing the implementation of 10 CFR 50, Appendix J, Option B for Types A, B and C tests. Option B also requires that a general visual inspection for structural deterioration of the

accessible interior and exterior surfaces of the containment system be conducted prior to each Type A test and at a periodic interval between tests, based on the performance of the containment system. These surfaces may affect the leak tight integrity of containment.

Section V.B.3 of Option B requires that the regulatory guide or other implementation document a licensee uses to develop a performance-based leakage testing program must be generally referenced in the TSs for the plant. Furthermore, the submittal for TS 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.

The implementation document that is currently referenced in TS 5.5.12 is Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Revision 0 of NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995, is endorsed by Regulatory Guide 1.163 as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B, subject to four regulatory positions delineated in Section C of Regulatory Guide 1.163. Revision 0 of NEI 94-01 includes provisions that allow the performance-based Type A test interval to be extended to up to 10 years, based upon two consecutive successful tests.

Revision 2-A of NEI 94-01 describes an approach for implementing the optional performance-based requirements of Option B. It incorporates the regulatory positions stated in Regulatory Guide 1.163 and includes provisions for extending Type A test intervals to up to 15 years. In the NRC final safety evaluation for Revision 2 of NEI 94-01, dated June 25, 2008, the NRC staff concluded that Revision 2-A of NEI 94-01 describes an acceptable approach for implementing the optional performance-based requirements of Option B and is acceptable for referencing by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to the specific limitations and conditions listed in Section 4.0 of the final safety evaluation for Revision 2 of NEI 94-01.

Revision 3 of NEI 94-01 includes guidance for extending Type C LLRT surveillance intervals up to 75 months. In the NRC final safety evaluation for Revision 3 of NEI 94-01, dated June 8, 2012, the NRC staff concluded that Revision 3 of NEI 94-01 describes an acceptable approach for implementing the optional performance-based requirements of Option B, as modified by the conditions and limitations summarized in Section 4.0 of the safety evaluation. The NRC staff also stated in the safety evaluation that the guidance, as modified to include two limitations and conditions, is acceptable for referencing by licensees proposing to amend their TSs in regard to containment

leakage rate testing. Revision 3-A of NEI 94-01 and Revision 2-A of NEI 94-01 include their corresponding NRC staff safety evaluations.

The proposed license amendment would revise TS 5.5.12 to follow the NRC approved guidance of NEI 94-01, Revision 3-A, with conditions and limitations specified in NEI 94-01, Revision 2-A, and delete two of the four listed exceptions to program guidelines, since they are no longer necessary. Based on the foregoing, the proposed license amendment would continue to ensure compliance with 10 CFR 50.54(o), and Option B of 10 CFR 50, Appendix J.

4.3 Conclusion

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 a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. 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. NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," July 1995, (ADAMS Legacy Library Accession Number 9510200180).
2. Order EA-12-050, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents," dated March 12, 2012 (Accession Number ML12054A694).
3. APC 17-13, "NRC Acceptance of Industry Guidance on Closure of PRA Peer Review Findings," dated May 8, 2017, with Attachment Appendix X.

4. NEI 05-04, “Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard,” November 2009.

5. NEI 07-12, “Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines,” June 2010.

6. NEI 12-13, “External Hazards PRA Peer Review Process Guidelines,” August 2012.

7. NRC Letter Issuing Amendment No 171 to Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit No. 1 - Issuance of Amendment Concerning Adoption of TSTF-425, “Relocate Surveillance Frequencies to Licensee Control” dated February 23, 2016 (Accession Number ML15307A349).

8. Generic Letter 88-20, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f),” Supplement 4, dated June 28, 1991.

9. Letter PY-CEI/NRR-2077L, “Resubmittal of the Individual Plant Examination - External Events,” dated July 22, 1996.

10. Letter from D.V. Pickett (NRC) to J.K. Wood (FENOC), “Perry Nuclear Power Plant, Unit 1 – Review of Individual Plant Examination of External Events (IPEEE) Submittal (TAC No. M83659),” dated March 9, 2001.

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

ATTACHMENT 1

Proposed Technical Specification Changes

(2 Pages follow)

5.5 Programs and Manuals

5.5.10 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.11 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases for these TS.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. a change in the TS incorporated in the license: or

   2. a change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

d. Proposed changes that meet the criteria of Specification 5.5.11.b.1 or Specification 5.5.11.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 Primary Containment Leakage Rate Testing Program

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

(continued)

NEI Topical Report NEI 94-01, Revision 3-A, with conditions and limitations in NEI 94-01, Revision 2-A,

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

- BN-TOP-1 methodology may be used for Type A tests.

- ~~The corrections to NEI 94-01 which are identified on the Errata Sheet attached to the NEI letter, "Appendix J Workshop Questions and Answers," dated March 19, 1996 are considered an integral part of NEI 94-01.~~

- The containment isolation check valves in the Feedwater penetrations are tested per the INSERVICE TESTING PROGRAM.

- ~~The provisions of NEI 94-01, Section 9.2.3 are revised to include the following exception: The first Type A test performed after the Type A test completed on July 1, 1994 shall be completed no later than June 29, 2009.~~

The peak calculated primary containment internal pressure for the design basis loss of coolant accident is 6.40 psig. For conservatism P_a is defined as 7.80 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.20% of primary containment air weight per day at the peak containment pressure (P_a).

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup following testing performed in accordance with this Program, the leakage rate acceptance criteria are $< 0.6\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 2.5 scfh when tested at $\geq P_a$.

   2) For each door, leakage rate is ≤ 2.5 scfh when the gap between the door seals is pressurized to $\geq P_a$.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing frequencies required by 10 CFR 50, Appendix J.

(continued)

ATTACHMENT 2

Evaluation of Risk Significance of Permanent ILRT Extension

(39 Pages follow)

JENSEN HUGHES

Advancing the Science of Safety

Perry Nuclear Power Plant:
Evaluation of Risk Significance of Permanent ILRT Extension

54008-CALC-01

Prepared for:
Perry Nuclear Power Plant

Project Title: Permanent ILRT Extension

Revision: 1

		Name and Date
Preparer: Justin Sattler		
Reviewer: Matthew Johnson		
Reviewer: Kelly Wright		
Review Method	Design Review ☒ Alternate Calculation ☐	
Approved by: Richard Anoba		

REVISION RECORD SUMMARY

Revision	Revision Summary
0	Initial Issue with comments incorporated
1	Updated dose calculation (Section 5.2.2) and PRA Technical Adequacy (Attachment 1)

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this analysis is to provide a risk assessment of permanently extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the Perry Nuclear Power Plant (PNPP). The risk assessment follows the guidelines from NEI 94-01, Revision 3-A [Reference 1], the methodology used in EPRI TR-104285 [Reference 2], the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 [Reference 3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in Regulatory Guide 1.200 as applied to ILRT interval extensions, risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [Reference 4], 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 [Reference 5], and the methodology used in EPRI 1018243, Revision 2-A of EPRI 1009325 [Reference 24].

2.0 SCOPE

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing frequency requirement from three in ten years to at least once in ten years. The revised Type A 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 limiting containment leakage rate of $1L_a$.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and 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 [Reference 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 assessment 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 TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals."

The NRC report on performance-based leak testing, NUREG-1493, 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 PWR plant (i.e., Surry), that containment isolation failures contribute less than 0.1% to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for PNPP.

NEI 94-01 Revision 3-A supports using EPRI Report No. 1009325 Revision 2-A (EPRI 1018243), "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," for performing risk impact assessments in support of ILRT extensions [Reference 24]. The Guidance provided in Appendix H of EPRI Report No. 1009325 Revision 2-A builds on the EPRI Risk Assessment methodology, EPRI TR-104285. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT change.

It should be noted that containment leak-tight integrity is also verified through periodic in-service 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 in-service 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. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E) require licensees to conduct visual inspections of the accessible areas of the interior of the containment. The associated change to NEI 94-01 will require that visual examinations be conducted during at least three other outages, and in the outage during which the ILRT is being conducted. These requirements will not be changed as a result of the extended ILRT 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 are also not affected by the change to the Type A test frequency.

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this permanent 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 10^{-6} per reactor year and increases in Large Early Release Frequency (LERF) less than 10^{-7} per reactor year. Perry does not credit containment overpressure for NPSH for ECCS. Therefore, the Type A test does not impact CDF, so the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-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), which helps ensure 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 ILRT intervals. In context, it is noted that a CCFP of 1/10 (10%) has been approved for application to evolutionary light water designs. Given these perspectives, a change in the CCFP of up to 1.5% is assumed to be small.

In addition, the total annual risk (person rem/year population dose) is examined to demonstrate the relative change in this parameter. While no acceptance guidelines for these additional figures of merit are published, examinations of NUREG-1493 and Safety Evaluation Reports (SER) for one-time interval extension (summarized in Appendix G of Reference 24) indicate a range of incremental increases in population dose that have been accepted by the NRC. The range of incremental population dose increases is from ≤0.01 to 0.2 person-rem/year and/or 0.002% to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (NUREG-1493 [Reference 6], Figure 7-2) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a very small population dose is defined as an increase from the baseline interval (3 tests per 10 years) dose of ≤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 ILRT interval.

3.0 REFERENCES

The following references were used in this calculation:

1. *Revision 3-A to Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J*, NEI 94-01, July 2012.
2. *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, EPRI, Palo Alto, CA EPRI TR-104285, August 1994.
3. *Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, Revision 4, developed for NEI by EPRI and Data Systems and Solutions, October 2001.
4. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, Revision 3, January 2018.
5. *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension*, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.
6. Performance-Based Containment Leak-Test Program, NUREG-1493, September 1995.
7. *Evaluation of Severe Accident Risks: Methodology for the Containment, Source Term, Consequence, and Risk Integration Analyses*, NUREG/CR-4551, SAND86-1309, Volume 4, Revision 1, Part 1, December 1990.
8. Letter from R. J. Barrett (Entergy) to U. S. Nuclear Regulatory Commission, IPN-01-007, January 18, 2001.
9. United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.
10. *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.
11. *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.
12. Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check', NUREG-1273, April 1988.
13. *Review of Light Water Reactor Regulatory Requirements*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Volume 2, June 1986.
14. Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™, EPRI, Palo Alto, CA, TR-105189, Final Report, May 1995.
15. *Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants*, NUREG-1150, December 1990.
16. United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.
17. QU-001, Revision 1, PRA-PY1-FP-R1, Perry Nuclear Power Plant, "Quantification Notebook," April 2014.

18. L2-001, Revision 1, PRA-PY1-FP-R1, Perry Nuclear Power Plant, "Level 2 Notebook," April 2014.
19. *Evaluation of Severe Accident Risks: Grand Gulf, Unit 1*, NUREG/CR-4551, SAND86-1309, Volume 6, Revision 1, Parts 1 and 2, December 1990.
20. Anthony R. Pietrangelo, One-time extensions of containment integrated leak rate test interval – additional information, NEI letter to Administrative Points of Contact, November 30, 2001.
21. Letter from J. A. Hutton (Exelon, Peach Bottom) to U. S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.
22. *Risk Assessment for Joseph M. Farley Nuclear Plant Regarding ILRT (Type A) Extension Request*, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, P0293010002-1929-030602, March 2002.
23. Letter from D. E. Young (Florida Power, Crystal River) to U. S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.
24. *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, Revision 2-A of 1009325, EPRI, Palo Alto, CA. 1018243, October 2008.
25. Risk Assessment for Vogtle Electric Generating Plant Regarding the ILRT (Type A) Extension Request, prepared for Southern Nuclear Operating Co. by ERIN Engineering and Research, February 2003.
26. Perspectives Gained from the IPEEE Program, USNRC, NUREG-1742, April 2002.
27. *Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME/ANS RA-Sa–2009, February 2009.
28. ML112070867, "Containment Liner Corrosion Operating Experience Summary Technical Letter Report," U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, August 2011.
29. Technical Letter Report ML112070867, Containment Liner Corrosion Operating Experience Summary, Revision 1, August 2011.
30. Letter PY-CEI/NRR-2077L "Re-submittal of the Individual Plant Examination – External Events," dated July 22, 1996.
31. PRA-PY1-FP-R1, DR-001, Documentation Roadmap Notebook.
32. NOPM-CC-6000, "Probabilistic Risk Assessment Program," Revision 5.
33. NOBP-CC-6001, "Probabilistic Risk Assessment Model Management," Revision 7.
34. *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Regulatory Guide 1.200, Revision 2, March 2009.
35. ML14083A586, EPRI Evaluation, "Fleet Seismic Core Damage Frequency Estimates for Central and Eastern U.S. Nuclear Power Plants Using New Site-Specific Seismic Hazard Estimates," March 11, 2014.
36. Generic Issue 199 (GI-199), ML100270582, September 2010, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment."

37. NRC Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities -10 CFR 50.54(f), Supplement 4," June 28, 1991.

38. NUREG-1407, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991.

39. Letter form Douglas V. Pickett (USNRC) to Mr. John K. Wood, FirstEnergy Nuclear Operating Company subject "Perry Nuclear Power Plant, Unit 1 – Review of Individual Plant Examination of External Events (IPEEE) Submittal (TAC NO. M83659)" dated March 9, 2001. Docket No. 50-440

40. Updated Safety Analysis Report (USAR), Revision 19, October 2015.

41. Letter L-15-235, Perry Nuclear Power Plant, "Response to Request For Additional Information Regarding License Amendment to Adopt Technical Specification Task Force Traveler-425 (TAC No. MF3720)," ML15237A035, August 24, 2015.

42. Preliminary Notification of Event or Unusual Occurrence PNO-1-06-012, Evaluation of Oyster Creek Containment Design Function, ML063130424, November 9, 2006.

43. Surveillance SVI-T23-T0400, Drywell Leak Rate Test, July 1994.

44. Letter L-14-106, Perry Nuclear Power Plant, Unit No. 1 – Issuance of Amendment Concerning Adoption of TSTF-425, "Relocate Surveillance Frequencies to Licensee Control" (CAC No. MF3720), February 23, 2016.

45. ILRT-R-2009-04, Perry Nuclear Power Plant, Reactor Containment Building Integrated Leakage Rate Test Report, May 3-4, 2009.

46. APC 17-13, Subject: "NRC Acceptance of Industry Guidance on Closure of PRA Peer Review Findings," dated May 8th, 2017 with attachment Appendix X.

47. NEI 05-04, Revision 3, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Nuclear Energy Institute, November 2009.

48. ML17121A271, Subject: "US Nuclear Regulatory Commission Staff Expectations for an Industry Facts and Observations Independent Assessment Process," dated May 1st, 2017.

49. ML17095A252, Subject: "United States Nuclear Regulatory Commission Audit Report on Observation of Industry Independent Assessment Team Close-Out of Facts and Observations (F&Os)," dated May 1st, 2017.

50. NEI 07-12, Revision 1, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Nuclear Energy Institute, July 2010.

51. NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," August 2012.

52. PWROG-17074-P, Revision 0 "Perry Nuclear Power Plant Independent Assessment of Internal Events, Internal Flood, and Seismic PRA Facts & Observations Closure," PA-RMSC-1476, dated February 2018.

53. PWROG-17077-P, Rev. 0, "Perry Nuclear Power Plant Focused Scope Peer Review of Internal Events PRA and Seismic PRA," February 2018.

54. "PRA GAP ASSESSMENT RG 1.200 vs PRA for Perry Nuclear Power Plant," Rev. 0, Scientech Inc., July 2008

55. "Perry Nuclear Power Plant Evaluation of LERF Model against RG 1.200 and ASME PRA Standard," Rev. 0, Maracor, Inc., November 2011.

56. "Perry Nuclear Power Plant Internal Flood PRA Peer Review Report Using ASME PRA Standard Requirements," BWROG, August 2012.

57. ABS Consulting 3552894-R-002 Revision 2, "Perry Nuclear Power Plant Offsite Power Recovery PRA Peer Review Report Using ASME/ANS PRA Standard Requirements," August 4, 2015

58. Perry Nuclear Power Station Seismic PRA Peer Review Report Using ASME/ANS PRA Standard Requirements, November 2014.

4.0 ASSUMPTIONS AND LIMITATIONS

The following assumptions were used in the calculation:

- The technical adequacy of the PNPP PRA is consistent with the requirements of Regulatory Guide 1.200 [Reference 34] as is relevant to this ILRT interval extension, as detailed in Attachment 1.
- The PNPP Level 1 and Level 2 internal events PRA models provide representative results.
- It is appropriate to use the PNPP internal events PRA model to effectively describe the risk change attributable to the ILRT extension. An extensive sensitivity study is done in Section 5.3.1 to show the effect of including external event models for the ILRT extension. A Seismic PRA [Reference 36] and Fire PRA from the IPEEE [Reference 30] are used for this sensitivity analysis. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if detailed analysis of high wind events were to be included in the calculations.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [Reference 2].
- The representative containment leakage for Class 1 sequences is $1L_a$. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is $10L_a$ based on the previously approved methodology performed for Indian Point Unit 3 [Reference 8, Reference 9].
- The representative containment leakage for Class 3b sequences is $100L_a$ based on the guidance provided in EPRI Report No. 1009325, Revision 2-A (EPRI 1018243) [Reference 24].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [Reference 8, Reference 9].
- The impact on population doses from containment bypass scenarios is not altered by the proposed ILRT 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 in the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The drywell bypass leakage test (DBLT) is not being extended. The PRA model accounts for the DBLT. Therefore, extending the ILRT interval does not affect the DBLT.

5.0 METHODOLOGY AND ANALYSIS

5.1 Inputs

This section summarizes the general resources available as input (Section 5.1.1) and the plant specific resources required (Section 5.1.2).

5.1.1 General Resources Available

Various industry studies on containment leakage risk assessment are briefly summarized here:

1. NUREG/CR-3539 [Reference 10]
2. NUREG/CR-4220 [Reference 11]
3. NUREG-1273 [Reference 12]
4. NUREG/CR-4330 [Reference 13]
5. EPRI TR-105189 [Reference 14]
6. NUREG-1493 [Reference 6]
7. EPRI TR-104285 [Reference 2]
8. NUREG-1150 [Reference 15] and NUREG/CR-4551 [Reference 7]
9. NEI Interim Guidance [Reference 3, Reference 20]
10. Calvert Cliffs liner corrosion analysis [Reference 5]
11. EPRI Report No. 1009325, Revision 2-A (EPRI 1018243), Appendix H [Reference 24]

This first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PRA for the size of containment leakage that is considered significant and is to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and local leak rate test (LLRT) intervals on at-power public risk. The eighth study provides an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the basis for the consequence analysis of the ILRT interval extension for PNPP. The ninth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the ILRT interval. The tenth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. Finally, the eleventh study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with a permanent 15-year extension of the ILRT interval.

<u>NUREG/CR-3539 [Reference 10]</u>

Oak Ridge National Laboratory documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [Reference 16] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

<u>NUREG/CR-4220 [Reference 11]</u>

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to

calculate the unavailability of containment due to leakage.

NUREG-1273 [Reference 12]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [Reference 13]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

"...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189 [Reference 14]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because it provides insight regarding the impact of containment testing on shutdown risk. This study contains a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk. The conclusion from the study is that a small, but measurable, safety benefit is realized from extending the test intervals.

NUREG-1493 [Reference 6]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.

Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [Reference 2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures

4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failures due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

"...the proposed CLRT (Containment Leak Rate Tests) frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.04 person-rem per year..."

NUREG-1150 [Reference 15] and NUREG/CR-4551 [Reference 7]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec Leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Peach Bottom. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the PNPP Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent PNPP. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [Reference 3, Reference 20]

The guidance provided in this document builds on the EPRI risk impact assessment methodology [Reference 2] and the NRC performance-based containment leakage test program [Reference 6], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

Calvert Cliffs Response to Request for Additional Information Concerning the License Amendment for a One-Time Integrated Leakage Rate Test Extension [Reference 5]

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms was factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner.

EPRI Report No. 1009325, Revision 2-A, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals [Reference 24]

This report provides a generally applicable assessment of the risk involved in extension of ILRT test intervals to permanent 15-year intervals. Appendix H of this document provides guidance for performing plant-specific supplemental risk impact assessments and builds on the previous EPRI risk impact assessment methodology [Reference 2] and the NRC performance-based containment leakage test program [Reference 6], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the PNPP assessment to determine the estimated increase in risk associated with the ILRT extension. This 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 5.2.

5.1.2 Plant Specific Inputs

The plant-specific information used to perform the PNPP ILRT Extension Risk Assessment includes the following:

- Level 1 and LERF Model results [Reference 17]
- Level 2 Model results [Reference 18]
- Dose within a 50-mile radius [Reference 40]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [Reference 45]
- Containment failure probability data [Reference 18]

PNPP Model

The Internal Events PRA Model that is used for PNPP is characteristic of the as-built plant. The current Level 1, LERF, and Level 2 model (version PRA-PY1-FP-R1) is a linked fault tree model. The total CDF is 3.02E-6/year [Reference 17]. Table 5-1 and Table 5-2 provide a summary of the Internal Events CDF and LERF results for the PNPP PRA Model when quantified using the Level 1 model top gates. The total LERF is 1.39E-7/year.

The Fire CDF from the IPEEE [Reference 30] is 3.1E-5/year. The Seismic PRA results from Generic Issue 199 yields a CDF of 1.11E-5/year [Reference 36]. Refer to Section 5.3.1 for further details on external events as they pertain to this analysis.

Table 5-1 – Internal Events CDF

Internal Events	Frequency (per year)
Internal Floods	7.95E-07*
Transients	1.07E-06
LOOP	1.12E-06
ISLOCA	4.53E-09
LOCAs	2.23E-08
Total Internal Events CDF	3.02E-06

*Note: this Internal Flood CDF differs slightly from the value reported in Section 4.1 of Reference 17 (7.902E-7). This has no effect on the ILRT analysis results. This value is taken from totaling the % contributions in Table 18 of Reference 17 and is used to preserve total CDF.

Table 5-2 – Internal Events LERF

Internal Events	Frequency (per year)
Internal Floods	1.14E-07
Transients	1.36E-08
LOOP	6.57E-09
ISLOCA	4.44E-09
LOCAs	6.67E-12
Total Internal Events LERF	1.39E-07

Release Category Definitions

Table 5-3 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology [Reference 2]. 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 5.2 of this report.

Table 5-3 – EPRI Containment Failure Classification [Reference 2]

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 Individual Plant Examinations) including 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.
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 including 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 test and their potential failures.
6	Containment isolation failures including 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.

5.1.3 Impact of Extension on Detection of Component Failures that Lead to Leakage (Small and Large)

The ILRT 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 ILRT test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly addressed, the EPRI Class 3 accident class, as defined in Table 5-3, 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 Class 3b failures is determined consistent with the EPRI Guidance [Reference 24]. 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 "large" failures in 217 tests (i.e., 2 / 217 = 0.0092). For Class 3b, the probability is based on the Jeffreys non-informative prior (i.e., 0.5 / 218 = 0.0023).

In a follow-up letter [Reference 20] to their ILRT guidance document [Reference 3], NEI issued additional information concerning the potential that the calculated ΔLERF values for several plants may fall above the "very small change" guidelines of the NRC Regulatory Guide 1.174 [Reference 4]. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for ΔLERF. NEI describes ways to demonstrate that, using plant-specific

calculations, the ΔLERF 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 PNPP, as detailed in Section 5.2, involves subtracting the LERF 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.

Consistent with the NEI Guidance [Reference 3], 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 years / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 years / 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 ILRT testing. Correspondingly, an extension of the ILRT interval to 15 years can be estimated to lead to a factor of 5 ((15/2)/1.5) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC [Reference 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 ILRT extension.

5.2 Analysis

The application of the approach based on the guidance contained in EPRI Report No. 1009325, Revision 2-A, Appendix H [Reference 24], EPRI TR-104285 [Reference 2] and previous risk assessment submittals on this subject [References 5, 8, 21, 22, and 23] have led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report, as described in Table 5-4.

The analysis performed examined PNPP-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the breakdown 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 TR-104285, Class 1 sequences [Reference 2]).
- 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 bellow leakage (EPRI TR-104285, Class 3 sequences [Reference 2]).
- Accident sequences involving containment bypass (EPRI TR-104285, Class 8 sequences [Reference 2]), large containment isolation failures (EPRI TR-104285, Class 2 sequences [Reference 2]), and small containment isolation "failure-to-seal" events

(EPRI TR-104285, Class 4 and 5 sequences [Reference 2]) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency 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 5-4 – EPRI Accident Class Definitions

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
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 baseline risk in terms of frequency per reactor year for each of the accident classes presented in Table 5-4.

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 in 10 years to 1 in 15 years and 1 in 10 years to 1 in 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [Reference 4].

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

5.2.1 Step 1 – Quantify the Baseline Risk in Terms of Frequency per Reactor Year

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 ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285 [Reference 2].) The question on containment integrity was 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 were 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 5-4 were developed for PNPP by first determining the frequencies for Classes 1, 2, 6, 7, and 8. Table 5-5 presents the release categories, frequency, and EPRI category for each sequence and the totals of each EPRI classification. Table 5-6 provides a summary of the accident sequence frequencies that can lead to radionuclide release to the public and have been derived consistent with the definitions of accident classes defined in EPRI TR-104285 [Reference 2], the NEI Interim Guidance [Reference 3], and guidance provided in EPRI Report No. 1009325, Revision 2-A [Reference 24]. 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 5.2.6.

Note: calculations were performed with more digits than shown in this calculation. Therefore, minor differences may occur if the calculations in these sections are followed explicitly.

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 that can only be detected by performing a Type A ILRT. The probability of leakage detectable by a Type A ILRT is calculated to determine the impact of extending the testing interval. The Class 3 calculation is divided into two classes: Class 3a is defined as a small liner breach (L_a < leakage < $10L_a$), and Class 3b is defined as a large liner breach ($10L_a$ < leakage < $100L_a$).

Data reported in EPRI 1009325, Revision 2-A [Reference 24] states that two events could have been detected only during the performance of an ILRT and thus impact risk due to change in ILRT frequency. There were a total of 217 successful ILRTs during this data collection period. Therefore, the probability of leakage is determined for Class 3a as shown in the following equation:

$$P_{class3a} = \frac{2}{217} = 0.0092$$

Multiplying the CDF by the probability of a Class 3a leak yields the Class 3a frequency contribution in accordance with guidance provided in Reference 24. As described in Section 5.1.3, additional consideration is made to not apply failure probabilities on those cases that are already LERF scenarios. Therefore, these LERF contributions from CDF are removed. Therefore, the frequency of a Class 3a failure is calculated by the following equation:

$$Freq_{class3a} = P_{class3a} * (CDF - LERF) = \frac{2}{217} * (3.02\text{E-}6 - 1.39\text{E-}7) = 2.65\text{E-}8$$

In the database of 217 ILRTs, there are zero containment leakage events that could result in a large early release. Therefore, the Jeffreys non-informative prior is used to estimate a failure rate and is illustrated in the following equations:

$$\text{Jeffreys Failure Probability} = \frac{Number\ of\ Failures + 1/2}{Number\ of\ Tests + 1}$$

$$P_{class3b} = \frac{0 + 1/2}{217 + 1} = 0.0023$$

The frequency of a Class 3b failure is calculated by the following equation:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF)Intact = \frac{.5}{218} * (3.02\text{E-}6 - 1.39\text{E-}7) = 6.60\text{E-}9$$

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 Reference 24.

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). This frequency is determined by taking the risk from Level 2 cutsets that contain flags beginning with "FLAG-CI" (containment intact). The frequency per year is initially determined from the EPRI Accident Class 1 frequency listed in Table 5-5 and then subtracting the EPRI Class 3a and 3b frequency (to preserve total CDF), calculated below:

$$Freq_{class1} = Intact - (Freq_{class3a} + Freq_{class3b})$$

Class 2 Sequences. This group consists of core damage accident progression bins with large containment isolation failures. This frequency is determined by taking the risk from cutsets that contain flags beginning with "FLAG-EI." The frequency per year for these sequences is obtained from the EPRI Accident Class 2 frequency listed in Table 5-5.

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 ILRT, this group is not evaluated any further in the analysis, consistent with approved methodology.

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 ILRT, this group is not evaluated any further in this analysis, consistent with approved methodology.

Class 6 Sequences. 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. All other failure modes are bounded by the Class 2 assumptions. This accident class is also not evaluated further.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). This frequency is calculated by subtracting the Class 1, 2, and 8 frequencies from the total CDF. For this analysis, the frequency is determined from the EPRI Accident Class 7 frequencies listed in Table 5-5.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs, which includes interfacing system LOCAs (ISLOCAs) and breaks outside containment. This frequency is determined by taking the risk from cutsets that contain flags beginning with "IE-ISL" or "IE-BOC." For this analysis, the frequency is determined from the EPRI Accident Class 8 frequency listed in Table 5-5.

LERF quantification of the PRA-PY1-FP-R1 model [Reference 17] is distributed into EPRI classes based on release categories. Table 5-5 shows this distribution.

Table 5-5 – Accident Class Frequencies

Release Categories	EPRI Category	Frequency (/yr)
Intact Containment	Class 1	1.49E-06
Large Isolation Failure	Class 2	7.92E-08
Failures Induced by Phenomena	Class 7	1.44E-06
Bypass	Class 8	4.53E-09
Total (CDF)	N/A	3.02E-06

Table 5-6 – Baseline Risk Profile

Class	Description	Frequency (/yr)
1	No containment failure	1.46E-06[2]
2	Large containment isolation failures	7.92E-08
3a	Small isolation failures (liner breach)	2.65E-08
3b	Large isolation failures (liner breach)	6.60E-09
4	Small isolation failures - failure to seal (Type B)	ε[1]
5	Small isolation failures - failure to seal (Type C)	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]
7	Severe accident phenomena induced failure	1.44E-06

Table 5-6 – Baseline Risk Profile

Class	Description	Frequency (/yr)
8	Containment bypass	4.53E-09
	Total	3.02E-06

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
2. The Class 3a and 3b frequencies are subtracted from Class 1 to preserve total CDF.

5.2.2 Step 2 – Develop Plant-Specific Person-Rem Dose (Population Dose)

The population doses are calculated using the methodology of scaling Grand Gulf population doses to PNPP [Reference 24]. Comparison to Grand Gulf is used due to plant similarity (BWR-6 with a Mark III containment). The adjustment factor for reactor power level (AF_{power}) is defined as the ratio of the power level at PNPP (PLP) [Reference 40] to that at Grand Gulf Unit 1 (PLG) [Reference 19]. This adjustment factor is calculated as follows:

$AF_{power} = PLP / PLG = 3758 / 3833 = 0.980$

The adjustment factor for technical specification (TS) allowed containment leakage is defined as the ratio of the containment leakage at Perry (LRP) to that at Grand Gulf Unit 1 (LRG). This adjustment factor is calculated as follows:

$AF_{leakage} = LRP / LRG$

Since the leakage rates are in terms of the containment volume, the ratio of containment volumes is needed to relate the leakage rates. The TS maximum allowed containment leakage at PNPP (TS_P) is 0.2%/day [Table 1.3-4 of Reference 40]; the containment free volume at PNPP (VOL_P) is $1.1654*10^6$ ft^3 [Table 6.5-9 of Reference 40]. The TS maximum allowed containment leakage at Grand Gulf Unit 1 (TS_G) is 0.35%/day [Reference 40]; the containment free volume at Grand Gulf Unit 1 (VOL_G) is $1.40*10^6$ ft^3 [Reference 19]. Therefore,

$LRP = TS_P * VOL_P$

$LRG = TS_G * VOL_G$

$AF_{leakage} = (0.2 * 1.1654*10^6) / (0.35 * 1.40*10^6) = 0.476$

The adjustment factor for population ($AF_{Population}$) is defined as the ratio of the population within 50-mile radius of PNPP (POPP) [Reference 40] to that of Grand Gulf Unit 1 (POPG) [Reference 19]. Reference 40 provides 50-mile populations at several dates; for this calculation, the largest value (2,435,526) is used, so the calculation is conservative [Table 2.1-4 of Reference 40]. Reference 19 provides Grand Gulf populations at 1, 3, 10, 30, 100, 350, and 1000 miles from the plant. Since a 50-mile population is not provided, the 30-mile population (97,395) is conservatively used (a conservatively small Grand Gulf population yields a conservatively large multiplier for the doses used in this analysis). The population adjustment factor is calculated as follows:

$AF_{Population} = POPM / POPP = 2435526/ 97395= 25.01$

Consequences dependent on the INTACT TS Leakage (collapsed accident progression bins 8 and 10).

$AF_{INTACT} = AF_{power} * AF_{Leakage} * AF_{Population} = 0.980 * 0.476 * 25.01 = 11.66$

Since the other categories are not dependent on the TS Leakage, the adjustment factor (AF) is calculated by combining the factors as follows:

$AF = AF_{power} * AF_{Population} = 0.980 * 25.01 = 24.52$

The population dose data in NUREG/CR-4551 for Grand Gulf Unit 1 [Reference 19] is reported in nine distinct collapsed accident progression bins (CAPBs). For this ILRT extension application, CAPB7 and CAPB9 are categorized in EPRI Accident Class 1; CAPB3 is categorized in EPRI Accident Class 2; and CAPB6 is categorized in EPRI Accident Class 8. EPRI Accident Class 7 consists of a weighted average of CAPB3, CAPB4, and CAPB5 [Reference 24]. Based on the above adjustment factors and the 50-mile population dose (person-rem) for each CAPB considered in the NUREG/CR-4551 Grand Gulf Unit 1 study, the PNPP population doses (PPD) are calculated as follows:

$$PPD_{Class1} = AF_{INTACT} * PD_{CAPB7} + AF_{INTACT} * PD_{CAPB9} = 11.66 * 7.63E{+}2 + 11.66 * 0 = 8.90E{+}3$$

$$PPD_{Class2} = AF * PD_{CAPB3} = 24.52 * 2.00E{+}5 = 4.90E{+}6$$

$$PPD_{Class7} = AF * PD_{CAPB3,4,5} = 24.52 * 1.40E{+}5 = 3.43E{+}6$$

$$PPD_{Class8} = AF * PD_{CAPB6} = 24.52 * 1.30E{+}5 = 3.20E{+}6$$

Table 5-7 provides a correlation of PNPP population dose to EPRI Accident Class. Table 5-8 provides population dose for each EPRI accident class.

The population dose for EPRI Accident Classes 3a and 3b were calculated based on the guidance provided in EPRI Report No. 1009325, Revision 2-A [Reference 24] as follows:

$$EPRI\ Class\ 3a\ Population\ Dose = 10 * 8.90E{+}3 = 8.90E{+}4$$

$$EPRI\ Class\ 3b\ Population\ Dose = 100 * 8.90E{+}3 = 8.90E{+}5$$

Table 5-7 – Mapping of Population Dose to EPRI Accident Class

EPRI Category	Frequency (/yr)	Population Dose (person-rem)
Class 1	1.49E-06	8.90E+03
Class 2	7.92E-08	4.90E+06
Class 7	1.44E-06	3.43E+06
Class 8	4.53E-09	3.20E+06

Table 5-8 – Baseline Population Doses

Class	Description	Population Dose (person-rem)
1	No containment failure	8.90E+03
2	Large containment isolation failures	4.90E+06
3a	Small isolation failures (liner breach)	8.90E+04[1]
3b	Large isolation failures (liner breach)	8.90E+05[2]
4	Small isolation failures - failure to seal (type B)	N/A[3]
5	Small isolation failures - failure to seal (type C)	N/A[3]
6	Containment isolation failures (dependent failure, personnel errors)	N/A[3]
7	Severe accident phenomena induced failure	3.43E+06
8	Containment bypass	3.20E+06

1. $10*L_a$
2. $100*L_a$
3. The reason for the "N/A" is explained in Section 5.2.1.

5.2.3 Step 3 – Evaluate Risk Impact of Extending Type A Test Interval from 10 to 15 Years

The next step is to evaluate the risk impact of extending the test interval from its current 10-year interval to a 15-year interval. To do this, an evaluation must first be made of the risk associated with the 10-year interval, since the base case applies to 3-year interval (i.e., a simplified representation of a 3-to-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 Class 3b sequences is impacted. The risk contribution is changed based on the NEI guidance as described in Section 5.1.3 by a factor of 10/3 compared to the base case values. The Class 3a and 3b frequencies are calculated as follows:

$$Freq_{Class3a10yr} = \frac{10}{3} * \frac{2}{217} * (CDF - LERF) = \frac{10}{3} * \frac{2}{217} * 2.88\text{E-}6 = 8.85\text{E-}8$$

$$Freq_{Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * (CDF - LERF) = \frac{10}{3} * \frac{.5}{218} * 2.88\text{E-}6 = 2.20\text{E-}8$$

The results of the calculation for a 10-year interval are presented in Table 5-9.

Table 5-9 – Risk Profile for Once in 10 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	1.38E-06	45.72%	8.90E+03	1.23E-02
2	Large containment isolation failures	7.92E-08	2.62%	4.90E+06	3.88E-01
3a	Small isolation failures (liner breach)	8.85E-08	2.93%	8.90E+04	7.87E-03
3b	Large isolation failures (liner breach)	2.20E-08	0.73%	8.90E+05	1.96E-02
4	Small isolation failures - failure to seal (type B)	ε[1]	ε[1]	ε[1]	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]	ε[1]	ε[1]	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure	1.44E-06	47.84%	3.43E+06	4.95E+00
8	Containment bypass	4.53E-09	0.15%	3.20E+06	1.45E-02
	Total	3.02E-06			5.39E+00

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
2. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.

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 a factor of 5 compared to the 3-year interval value, as described in Section 5.1.3. The Class 3a and 3b frequencies are calculated as follows:

$$Freq_{Class3a15yr} = \frac{15}{3} * \frac{2}{217} * (CDF - LERF) = 5 * \frac{2}{217} * 2.88\text{E-}6 = 1.33\text{E-}7$$

$$Freq_{Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * (CDF - LERF) = 5 * \frac{.5}{218} * 2.88\text{E-}6 = 3.30\text{E-}8$$

The results of the calculation for a 15-year interval are presented in Table 5-10.

Table 5-10 – Risk Profile for Once in 15 Year ILRT

Class	Description	Frequency (/yr)	Contribution (%)	Population Dose (person-rem)	Population Dose Rate (person-rem/yr)
1	No containment failure[2]	1.32E-06	43.89%	8.90E+03	1.18E-02
2	Large containment isolation failures	7.92E-08	2.62%	4.90E+06	3.88E-01
3a	Small isolation failures (liner breach)	1.33E-07	4.40%	8.90E+04	1.18E-02
3b	Large isolation failures (liner breach)	3.30E-08	1.09%	8.90E+05	2.94E-02
4	Small isolation failures - failure to seal (type B)	ε[1]	ε[1]	ε[1]	ε[1]
5	Small isolation failures - failure to seal (type C)	ε[1]	ε[1]	ε[1]	ε[1]
6	Containment isolation failures (dependent failure, personnel errors)	ε[1]	ε[1]	ε[1]	ε[1]
7	Severe accident phenomena induced failure	1.44E-06	47.84%	3.43E+06	4.95E+00
8	Containment bypass	4.53E-09	0.15%	3.20E+06	1.45E-02
	Total	3.02E-06			5.40E+00

1. ε represents a probabilistically insignificant value or a Class that is unaffected by the Type A ILRT.
2. The Class 1 frequency is reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.

5.2.4 Step 4 – Determine the Change in Risk in Terms of LERF and Dose

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from 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 [Reference 4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 [Reference 4] defines very small changes in risk as resulting in increases of CDF less than 10^{-6}/year and increases in LERF less than 10^{-7}/year, and small changes in LERF as less than 10^{-6}/year. Since containment overpressure is not required in support of ECCS performance to mitigate design basis accidents

at PNPP, the ILRT extension does not impact CDF. Therefore, the relevant risk-impact metric is LERF.

For PNPP, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology). Based on a 3-in-10-year test interval from Table 5-8, the Class 3b frequency is 6.60E-9; based on a 10-year test interval from Table 5-9, the Class 3b frequency is 2.20E-8/year; based on a 15-year test interval from Table 5-10, the Class 3b frequency is 3.30E-8/year. Thus, the increase in LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 2.64E-8/year. Similarly, the increase due to increasing the interval from 10 to 15 years is 1.10E-8/year. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF is less than the threshold criteria for a very small change when comparing the 15-year results to the current 10-year requirement and the original 3-year requirement. Table 5-11 summarizes these results.

Table 5-11 – Impact on LERF due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	6.60E-09	2.20E-08	3.30E-08
ΔLERF (3 year baseline)		1.54E-08	2.64E-08
ΔLERF (10 year baseline)			1.10E-08

EPRI Report No. 1009325, Revision 2-A [Reference 24] states that a very small population dose is defined as an increase of ≤ 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 ILRT intervals. As shown in Table 5-12, the results of this calculation meet the dose rate criteria.

Table 5-12 – Impact on Dose Rate due to Extended Type A Testing Intervals

ILRT Inspection Interval	10 Years	15 Years
ΔDose Rate (3 year baseline)	1.853E-02	3.176E-02
ΔDose Rate (10 year baseline)		1.323E-02
%ΔDose Rate (3 year baseline)	0.345%	0.591%
%ΔDose Rate (10 year baseline)		0.246%

5.2.5 Step 5 – Determine the Impact on the Conditional Containment Failure Probability

Another parameter that the NRC guidance in RG 1.174 [Reference 4] states can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The CCFP is defined as the probability of containment failure given the occurrence of an accident. This probability can be expressed using the following equation:

$$CCFP = 1 - \frac{f(ncf)}{CDF}$$

where *f*(ncf) is the frequency of those sequences that do not result in containment failure; this frequency is determined by summing the Class 1 and Class 3a results.

Table 5-13 shows the steps and results of this calculation.

Table 5-13 – Impact on CCFP due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	1.48E-06	1.47E-06	1.46E-06
f(ncf)/CDF	0.492	0.487	0.483
CCFP	0.508	0.513	0.517
ΔCCFP (3 year baseline)		0.511%	0.875%
ΔCCFP (10 year baseline)			0.365%

As stated in Section 2.0, a change in the CCFP of up to 1.5% is assumed to be small. The increase in the CCFP from the 3 in 10 year interval to 1 in 15 year interval is 0.875%. Therefore, this increase is judged to be very small.

5.2.6 Impact of Extension on Detection of Steel Liner Corrosion that Leads to Leakage

An estimate of the likelihood and risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is evaluated using a methodology similar to the Calvert Cliffs liner corrosion analysis [Reference 5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner.

The following approach is used to determine the change in likelihood, due to extending the ILRT, 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 the containment cylinder and dome
- 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

<u>Assumptions</u>

- Based on a review of industry events, an Oyster Creek incident is assumed to be applicable to PNPP for a concealed shell failure in the floor. In the Calvert Cliffs analysis, this event was assumed not to be applicable and a half failure was assumed for basemat concealed liner corrosion due to the lack of identified failures (See Table 5-14, Step 1).
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs previous analysis are assumed to still be applicable.
- 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) (See Table 5-14, 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 5-14, 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. For PNPP, the containment median failure pressure is 64 psig [Reference 18]. Probabilities of 1% for the cylinder and dome, and 0.1% for the basemat are used in this analysis, and sensitivity studies are included in Section 5.3.2 (See Table 5-14, 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 5-14, 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 5-14, Step 5).
- 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 5-14 – Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome	Containment Basemat
1	Historical liner flaw likelihood Failure data: containment location specific Success data: based on 70 steel-lined containments and 5.5 years since the 10CFR 50.55a requirements of periodic visual inspections of containment surfaces	Events: 2 (Brunswick 2 and North Anna 2) 2 / (70 x 5.5) = 5.19E-03	Events: 1 (Oyster Creek) 1 / (70 x 5.5) = 2.60E-03
2	Aged adjusted liner flaw likelihood During the 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the 5th to 10th year set to the historical failure rate.	Year / Failure rate 1 / 2.05E-03 average 5-10 / 5.19E-03 15 / 1.43E-02 15 year average = 6.44E-03	Year / Failure rate 1 / 1.03E-03 average 5-10 / 2.60E-03 15 / 7.14E-03 15 year average = 3.22E-03
3	Increase in flaw likelihood between 3 and 15 years Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years.	0.73% (1 to 3 years) 4.18% (1 to 10 years) 9.66% (1 to 15 years)	0.36% (1 to 3 years) 2.08% (1 to 10 years) 4.82% (1 to 15 years)
4	Likelihood of breach in containment given liner flaw	1%	0.1%
5	Visual inspection detection failure likelihood	10% 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT). All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	100% Cannot be visually inspected
6	Likelihood of non-detected containment leakage (Steps 3 x 4 x 5)	0.00073% (3 years) 0.73% x 1% x 10% 0.00418% (10 years) 4.18% x 1% x 10% 0.00966% (15 years) 9.66% x 1% x 10%	0.000360% (3 years) 0.36% x 0.1% x 100% 0.00208% (10 years) 2.08% x 0.1% x 100% 0.00482% (15 years) 4.82% x 0.1% x 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 PNPP.

Table 5-15 – Total Likelihood on Non-Detected Containment Leakage Due to Corrosion for PNPP

Description
At 3 years: 0.00073% + 0.000360% = 0.00109%
At 10 years: 0.00418% + 0.00208% = 0.00626%
At 15 years: 0.00966% + 0.00482% = 0.01448%

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF.

The two corrosion events that were initiated from the non-visible (backside) portion of the containment liner used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to this containment analysis. These events, one at North Anna Unit 2 (September 1999) caused by timber embedded in the concrete immediately behind the containment liner, and one at Brunswick Unit 2 (April 1999) caused by a cloth work glove embedded in the concrete next to the liner, were initiated from the nonvisible (backside) portion of the containment liner. A search of the NRC website LER database identified two additional events have occurred since the Calvert Cliffs analysis was performed and another event at a BWR screened in the Calvert Cliffs analysis as not applicable. In January 2000, a 3/16-inch circular through-liner hole was found at Cook Nuclear Plant Unit 2 caused by a wooden brush handle embedded immediately behind the containment liner. The other event occurred in April 2009, where a through-liner hole approximately 3/8-inch by 1-inch in size was identified in the Beaver Valley Power Station Unit 1 (BVPS-1) containment liner caused by pitting originating from the concrete side due to a piece of wood that was left behind during the original construction that came in contact with the steel liner. Two other containment liner through-wall hole events occurred at Turkey Point Units 3 and 4 in October 2010 and November 2006, respectively. However, these events originated from the visible side caused by the failure of the coating system, which was not designed for periodic immersion service, and are not considered to be applicable to this analysis. More recently, in October 2013, some through-wall containment liner holes were identified at BVPS-1, with a combined total area of approximately 0.395 square inches. The cause of these through-wall liner holes was attributed to corrosion originating from the outside concrete surface due to the presence of rayon fiber foreign material that was left behind during the original construction and was contacting the steel liner [References 28 and 29]. In the mid-1980s, Oyster Creek identified corrosion of the shell of the containment drywell in the sandbed region [Reference 42]. For risk evaluation purposes, these six total corrosion events occurring in 66 operating plants with steel containment liners over a 17.1 year period from September 1996 to October 4, 2013 (i.e., 6/(66*17.1) = 5.32E-03) are bounded by the estimated historical flaw probability based on the three events in the 5.5 year period of the Calvert Cliffs analysis (i.e., 3/(70*5.5) = 7.79E-03) incorporated in the EPRI guidance.

5.3 Sensitivities

5.3.1 Potential Impact from External Events Contribution

An assessment of the impact of external events is performed. The primary purpose for this investigation is the determination of the total LERF following an increase in the ILRT testing interval from 3 in 10 years to 1 in 15 years.

The IPEEE Fire PRA calculated a CDF of 3.10E-5 [Reference 30]. Since no Fire LERF value is calculated, it is assumed the LERF/CDF ratio will be similar for fire risk as for internal events risk. Applying the internal event LERF/CDF ratio to the Fire CDF yields an estimated Fire LERF of 1.43E-6, as shown by the equation below.

$$LERF_{Fire} \approx CDF_{Fire} * LERF_{IE} / CDF_{IE} = 3.10E\text{-}5 * 1.39E\text{-}7 / 3.02E\text{-}6 = 1.43E\text{-}6$$

As described in Section 5.1.3, additional consideration is made to not apply failure probabilities on those cases that are already LERF scenarios. Therefore, these LERF contributions from CDF are removed. To reduce conservatism in the ILRT extension analysis, the methodology of

subtracting existing LERF from CDF is also applied to the Fire PRA model. The following shows the calculation for Class 3b:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (3.10E\text{-}5 - 1.43E\text{-}6) = 6.78\text{E-}8$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (3.10E\text{-}5 - 1.43E\text{-}6) = 2.26\text{E-}7$$

$$Freq_{U1class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (3.10E\text{-}5 - 1.43E\text{-}6) = 3.39\text{E-}7$$

The IPEEE Seismic Margin Assessment (SMA) does not result in an estimate of CDF [Reference 30]. There is also a Seismic PRA under development, but it is not an effective model because it does not reflect the as-built, as-operated facility due to plant procedures and/or modifications not being in place or currently under active development [References 52 and 53]; see Section A.2.2 for details. Since neither the IPEEE SMA nor the Seismic PRA under development provide an accurate risk quantification, another analysis is used.

The conclusions reached in 2010 by GI-199 [Reference 36] are used for estimating Seismic CDF at plants in the Central and Eastern United States, which includes PNPP. EPRI guidance [Reference 35] on recent seismic evaluations states, "EPRI does not recommend using any very conservative approaches to estimate the SCDF such as use of the maximum SCDFs calculated at any one frequency. This type of bounding approach is overly conservative and judged to not provide realistic risk estimates consistent with SCDFs calculated in actual SPRAs." Therefore, the average of the Seismic CDF values reported in Table D-1 of GI-199 [Reference 36] is calculated as follows:

$$\text{CDF}_{\text{Seismic}} = (2.1\text{E-}5 + 1.5\text{E-}5 + 6.4\text{E-}6 + 1.8\text{E-}6)/4 = 1.11\text{E-}5$$

Applying the internal event LERF/CDF ratio to the Seismic CDF yields an estimated seismic LERF of 5.09E-7, as shown by the equation below.

$$\text{LERF}_{\text{Seismic}} \approx \text{CDF}_{\text{Seismic}} * \text{LERF}_{\text{IE}} / \text{CDF}_{\text{IE}} = 1.11\text{E-}5 * 1.39\text{E-}7 / 3.02\text{E-}6 = 5.09\text{E-}7$$

Again subtracting LERF from CDF, the Class 3b frequency can be calculated by the following formulas:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (1.11\text{E-}5 - 5.09\text{E-}7) = 2.42\text{E-}8$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (1.11\text{E-}5 - 5.09\text{E-}7) = 8.06\text{E-}8$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = \frac{15}{3} * \frac{0.5}{218} * (1.11\text{E-}5 - 5.09\text{E-}7) = 1.21\text{E-}7$$

The IPEEE [Section 1.4.3 of Reference 30] states, "The review of the impact on PNPP of high winds, floods and other external hazards resulted in the conclusion that there are no significant events of concern." Therefore, for this sensitivity a conservative CDF of 10^{-6} is used for all other external events. Applying the internal event LERF/CDF ratio to the other external events CDF yields an estimated other external events LERF of 4.60E-8. Again subtracting LERF from CDF, the Class 3b frequency can be calculated by the following formulas:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (1.00\text{E-}6 - 4.60\text{E-}8) = 2.19\text{E-}9$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (1.00\text{E-}6 - 4.60\text{E-}8) = 7.29\text{E-}9$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = \frac{15}{3} * \frac{0.5}{218} * (1.00\text{E-}6 - 4.60\text{E-}8) = 1.09\text{E-}8$$

The fire, seismic, and other external events contributions to Class 3b frequencies are then combined to obtain the total external event contribution to Class 3b frequencies. The change in LERF is calculated for the 1 in 10 year and 1 in 15 year cases and the change defined for the external events in Table 5-16.

Table 5-16 – PNPP External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 3 per 10 years to 1 per 15 years)
	3 per 10 year	1 per 10 year	1 per 15 years	
External Events	9.42E-08	3.14E-07	4.71E-07	3.77E-07
Internal Events	6.60E-09	2.20E-08	3.30E-08	2.64E-08
Combined	1.01E-07	3.36E-07	5.04E-07	4.03E-07

The internal event results are also provided to allow a composite value to be defined. When both the internal and external event contributions are combined, the total change in LERF of 4.03E-7 meets the guidance for small change in risk, as it exceeds 1.0E-7/yr and remains less than 1.0E-6 change in LERF. For this change in LERF to be acceptable, total LERF must be less than 1.0E-5. The total LERF value is calculated below:

$$\begin{aligned} \mathrm{LERF} &= \mathrm{LERF}_{\mathrm{internal}} + \mathrm{LERF}_{\mathrm{fire}} + \mathrm{LERF}_{\mathrm{seismic}} + \mathrm{LERF}_{\mathrm{other}} + \mathrm{LERF}_{\mathrm{class3Bincrease}} \\ &= 1.39\mathrm{E}\text{-}7/\mathrm{yr} + 1.43\mathrm{E}\text{-}6/\mathrm{yr} + 5.09\mathrm{E}\text{-}7/\mathrm{yr} + 4.60\mathrm{E}\text{-}8/\mathrm{yr} + 4.03\mathrm{E}\text{-}7/\mathrm{yr} = 2.52\mathrm{E}\text{-}6/\mathrm{yr} \end{aligned}$$

As specified in Regulatory Guide 1.174 [Reference 4], since the total LERF is less than 1.0E-5, it is acceptable for the ΔLERF to be between 1.0E-7 and 1.0E-6.

5.3.2 Potential Impact from Steel Liner Corrosion Likelihood

A quantitative assessment of the contribution of steel liner corrosion likelihood impact was performed for the risk impact assessment for extended ILRT intervals. As a sensitivity run, the internal event CDF was used to calculate the Class 3b frequency. The impact on the Class 3b frequency due to increases in the ILRT surveillance interval was calculated for steel liner corrosion likelihood using the relationships described in Section 5.2.6. The EPRI Category 3b frequencies for the 3 per 10-year, 10-year, and 15-year ILRT intervals were quantified using the internal events CDF. The change in the LERF, change in CCFP, and change in Annual Dose Rate due to extending the ILRT interval from 3 in 10 years to 1 in 10 years, or to 1 in 15 years are provided in Table 5-17 – Table 5-19. The steel liner corrosion likelihood was increased by a factor of 1000, 10000, and 100000. Except for extreme factors of 10000 and 100000, the corrosion likelihood is relatively insensitive to the results.

Table 5-17 – Steel Liner Corrosion Sensitivity Case: 3B Contribution

	3b Frequency (3-per-10 year ILRT)	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	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)
Internal Event 3B Contribution	6.60E-09	2.20E-08	3.30E-08	1.54E-08	2.64E-08	1.10E-08
Corrosion Likelihood X 1000	6.68E-09	2.34E-08	3.78E-08	1.67E-08	3.11E-08	1.44E-08
Corrosion Likelihood X 10000	7.32E-09	3.58E-08	8.08E-08	2.85E-08	7.35E-08	4.50E-08
Corrosion Likelihood X 100000	1.38E-08	1.60E-07	5.11E-07	1.46E-07	4.97E-07	3.51E-07

Table 5-18 – Steel Liner Corrosion Sensitivity: CCFP

	CCFP (3-per-10 year ILRT)	CCFP (1-per-10 year ILRT)	CCFP (1-per-15 year ILRT)	CCFP Increase (3-per-10 to 1-per-10)	CCFP Increase (3-per-10 to 1-per-15)	CCFP Increase (1-per-10 to 1-per-15)
Baseline CCFP	5.08E-01	5.13E-01	5.17E-01	5.11E-03	8.75E-03	3.65E-03
Corrosion Likelihood X 1000	5.08E-01	5.14E-01	5.17E-01	5.16E-03	8.85E-03	3.69E-03
Corrosion Likelihood X 10000	5.09E-01	5.14E-01	5.18E-01	5.66E-03	9.71E-03	4.04E-03
Corrosion Likelihood X 100000	5.11E-01	5.21E-01	5.29E-01	1.07E-02	1.83E-02	7.62E-03

Table 5-19 –Steel Liner Corrosion Sensitivity: Dose Rate

	Dose Rate (3-per-10 year ILRT)	Dose Rate (1-per-10 year ILRT)	Dose Rate (1-per-15 year ILRT)	Dose Rate Increase (3-per-10 to 1-per-10)	Dose Rate Increase (3-per-10 to 1-per-15)	Dose Rate Increase (1-per-10 to 1-per-15)
Dose Rate	7.94E-03	2.65E-02	3.97E-02	1.85E-02	3.18E-02	1.32E-02
Corrosion Likelihood X 1000	8.03E-03	2.81E-02	4.55E-02	2.01E-02	3.74E-02	1.73E-02
Corrosion Likelihood X 10000	8.81E-03	4.30E-02	9.72E-02	3.42E-02	8.84E-02	5.42E-02
Corrosion Likelihood X 100000	1.66E-02	1.92E-01	6.15E-01	1.76E-01	5.98E-01	4.22E-01

5.3.3 Expert Elicitation Sensitivity

Another sensitivity case on the impacts of assumptions regarding pre-existing containment defect or flaw probabilities of occurrence and magnitude, or size of the flaw, is performed as described in Reference 24. In this sensitivity case, an expert elicitation was conducted to develop probabilities for pre-existing containment defects that would be detected by the ILRT only based on the historical testing data.

Using the expert knowledge, this information was extrapolated into a probability-versus-magnitude relationship for pre-existing containment defects. The failure mechanism analysis also used the historical ILRT data augmented with expert judgment to develop the results. Details of the expert elicitation process and results are contained in Reference 24. The expert elicitation process has the advantage of considering the available data for small leakage events, which have occurred in the data, and extrapolate those events and probabilities of occurrence to the potential for large magnitude leakage events.

The expert elicitation results are used to develop sensitivity cases for the risk impact assessment. Employing the results requires the application of the ILRT interval methodology using the expert elicitation to change the probability of pre-existing leakage in the containment.

The baseline assessment uses the Jeffreys non-informative prior and the expert elicitation sensitivity study uses the results of the expert elicitation. In addition, given the relationship between leakage magnitude and probability, larger leakage that is more representative of large early release frequency, can be reflected. For the purposes of this sensitivity, the same leakage magnitudes that are used in the basic methodology (i.e., 10 L_a for small and 100 L_a for large) are used here. Table 5-20 presents the magnitudes and probabilities associated with the Jeffreys non-informative prior and the expert elicitation used in the base methodology and this sensitivity case.

Table 5-20 – PNPP Summary of ILRT Extension Using Expert Elicitation Values (from Reference 24)

Leakage Size (L_a)	Expert Elicitation Mean Probability of Occurrence	Percent Reduction
10	3.88E-03	86%
100	2.47E-04	91%

Taking the baseline analysis and using the values provided in Tables 5-10 and 5-11 for the expert elicitation sensitivity yields the results in Table 5-21.

Table 5-21 – PNPP Summary of ILRT Extension Using Expert Elicitation Values

Accident Class	ILRT Interval							
	3 per 10 Years				1 per 10 Years		1 per 15 Years	
	Base Frequency	Adjusted Base Frequency	Dose (person-rem)	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)	Frequency	Dose Rate (person-rem/yr)
1	1.49E-06	1.48E-06	8.90E+03	1.32E-02	1.45E-06	1.29E-02	1.43E-06	1.27E-02
2	7.92E-08	7.92E-08	4.90E+06	3.88E-01	7.92E-08	3.88E-01	7.92E-08	3.88E-01
3a	N/A	1.12E-08	8.90E+04	9.94E-04	3.72E-08	3.31E-03	5.59E-08	4.97E-03
3b	N/A	7.11E-10	8.90E+05	6.33E-04	2.37E-09	2.11E-03	3.56E-09	3.16E-03
6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	1.44E-06	1.44E-06	3.43E+06	4.95E+00	1.44E-06	4.95E+00	1.44E-06	4.95E+00
8	4.53E-09	4.53E-09	3.20E+06	1.45E-02	4.53E-09	1.45E-02	4.53E-09	1.45E-02
Totals	3.02E-06	3.02E-06	N/A	5.37E+00	3.02E-06	5.37E+00	3.02E-06	5.37E+00
ΔLERF (3 per 10 yrs base)	N/A				1.66E-09		2.84E-09	
ΔLERF (1 per 10 yrs base)	N/A				N/A		1.19E-09	
CCFP	50.64%				50.69%		50.73%	

The results illustrate how the expert elicitation reduces the overall change in LERF and the overall results are more favorable with regard to the change in risk.

6.0 RESULTS

The results from this ILRT extension risk assessment for PNPP are summarized in Table 6-1.

Table 6-1 – ILRT Extension Summary

Class	Dose (person-rem)	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15 Years	
		Frequency	Person-Rem/Year	Frequency	Person-Rem/Year	Frequency	Person-Rem/Year
1	8.90E+03	1.46E-06	1.30E-02	1.38E-06	1.23E-02	1.32E-06	1.18E-02
2	4.90E+06	7.92E-08	3.88E-01	7.92E-08	3.88E-01	7.92E-08	3.88E-01
3a	8.90E+04	2.65E-08	2.36E-03	8.85E-08	7.87E-03	1.33E-07	1.18E-02
3b	8.90E+05	6.60E-09	5.87E-03	2.20E-08	1.96E-02	3.30E-08	2.94E-02
6	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	3.43E+06	1.44E-06	4.95E+00	1.44E-06	4.95E+00	1.44E-06	4.95E+00
8	3.20E+06	4.53E-09	1.45E-02	4.53E-09	1.45E-02	4.53E-09	1.45E-02
Total		**3.02E-06**	**5.37E+00**	**3.02E-06**	**5.39E+00**	**3.02E-06**	**5.40E+00**
ILRT Dose Rate from 3a and 3b							
ΔTotal Dose Rate	From 3 Years	N/A		1.85E-02		3.18E-02	
	From 10 Years	N/A		N/A		1.32E-02	
%ΔDose Rate	From 3 Years	N/A		0.345%		0.591%	
	From 10 Years	N/A		N/A		0.246%	
3b Frequency (LERF)							
ΔLERF	From 3 Years	N/A		1.54E-08		2.64E-08	
	From 10 Years	N/A		N/A		1.10E-08	
CCFP %							
ΔCCFP %	From 3 Years	N/A		0.511%		0.875%	
	From 10 Years	N/A		N/A		0.365%	

7.0 CONCLUSIONS AND RECOMMENDATIONS

Based on the results from Section 5.2 and the sensitivity calculations presented in Section 5.3, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test frequency to 15 years:

- Regulatory Guide 1.174 [Reference 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 less than 1.0E-06/year and increases in LERF less than 1.0E-07/year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years is estimated as 2.64E-8/year using the EPRI guidance; this value increases negligibly if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included. As such, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4].
- The effect resulting from changing the Type A test frequency to 1-per-15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.0318 person-rem/year. EPRI Report No. 1009325, Revision 2-A [Reference 24] states that a very small population dose is defined as an increase of ≤ 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 ILRT intervals. The results of this calculation meet these criteria. Moreover, the risk impact for the ILRT extension when compared to other severe accident risks is negligible.
- The increase in the conditional containment failure probability from the 3 in 10 year interval to 1 in 15 year interval is 0.875%. EPRI Report No. 1009325, Revision 2-A [Reference 24] states that increases in CCFP of ≤ 1.5% is very small. Therefore, this increase is judged to be very small.
- Multiple sensitivities are performed in Section 5.3. As shown in Section 5.3.1, when both the internal and external event contributions are combined, the total change in LERF of 4.03E-7 meets the guidance for small change in risk, as it exceeds 1.0E-7/yr and remains less than 1.0E-6 change in LERF and the total LERF is 2.52E-6. Other sensitivities show the baseline ILRT extension analysis (as performed in Section 5.2) is conservative.

Therefore, increasing the ILRT interval to 15 years is considered to be small since it represents a small change to the PNPP risk profile.

<u>Previous Assessments</u>

The NRC in NUREG-1493 [Reference 6] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from 3 per 10 years to 1 per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B or Type C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between

integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond 1 in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test integrity of the containment structure.

The conclusions for PNPP confirm these general conclusions on a plant-specific basis considering the severe accidents evaluated for PNPP, the PNPP containment failure modes, and the local population surrounding PNPP.

A. ATTACHMENT 1 – PRA TECHNICAL ADEQUACY

A.1. Internal Events PRA Quality Statement for Permanent 15-Year ILRT Extension

No open findings related to the internal events PRA model remain at PNPP.

The PNPP PRA underwent an independent assessment in accordance with the guidance of Appendix X [Reference 46] of Nuclear Energy Institute (NEI) 05-04, NEI 07-12, and NEI 12-13 [References 47, 50, and 51] as well as the expectations of the NRC [References 48]. The independent assessment also considered the lessons learned from the NRC staff observations (i.e., audits) of the three independent assessment team pilot reviews [Reference 49].

The final report [Reference 52] documents the review process conducted, the F&Os reviewed for Internal Events, Internal Flooding, and Large Early Release (LERF) models and the Seismic model, and the overall results. The independent assessment concluded that model upgrades occurred, and resulted in a focused scope peer review of common cause modeling and human reliability analysis [Reference 53]. No new findings resulted from that review, and ultimately the satisfaction (closure) of all outstanding findings identified in previous peer reviews associated with the PNPP Internal Events/Internal Flooding/LERF models was achieved. Previous reviews included a 2008 Gap Analysis Self-Assessment (Internal Events) against the PRA Standard (2005 revision) [Reference 54], a 2011 Focused Large Early Release Frequency Peer Review [Reference 55], and a 2012 Focused Internal Flood PRA Peer Review [Reference 56]. A focused scope peer review of the offsite power recovery modeling in 2015 did not result in any identified findings [Reference 57].

The PNPP Independent Assessment is documented as part of the PRA model process and is available for auditable purposes.

A.1.1 PRA Maintenance and Update

The PNPP PRA model of record and supporting documentation have been maintained as a living program, with updates directed every other refueling cycle (every four years) to reflect the as-built, as-operated plant. Interim updates may be prepared and issued in between regularly scheduled model updates on an as-needed basis. Typically, an interim revision would be used for an update that would cause a change in Core Damage Frequency of greater than 10 percent, a change in Large Early Release Frequency of greater than 20 percent, or for changes that could significantly impact a risk-informed application. Interim models are also released following focused peer reviews once the associated findings and suggestions have been addressed.

FENOC employs a multi-faceted, structured approach in establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all FENOC nuclear generation sites. This approach includes a procedure that documents the PRA maintenance and update process, as well as the use of self-assessments and independent peer reviews.

The FENOC risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plant. This process is defined in the FENOC PRA Program, which consists of a governing procedure [Reference 32] and subordinate implementation procedures. The governing procedure serves as the higher tier procedure and establishes the FENOC PRA Program and provides administrative requirements for the maintenance and upgrade of the FENOC PRA models and risk-informed applications. The overall objective of the PRA Program is to provide technically adequate PRA models such that the requirements set forth in Regulatory Guide 1.200, Revision 2, are satisfied for use in risk-

informed applications. PRA related documentation is maintained within the records capture process.

In addition to model control, administrative mechanisms are in place to assure that changes from plant modifications, procedure changes, changes to calculations, and industry operating experience are appropriately screened, dispositioned, and tracked for incorporation into the PRA model if that change would impact the model.

A.1.2 Consistency with Applicable PRA Standards

An internal gap assessment between ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" and ASME/ANS RA-Sb-2005, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" was performed in support of the TSTF-425 License Amendment Request (LAR) Request for Additional Information (RAI) responses [RAI 3 of Reference 44]. Between ASME/ANS RA-Sb-2005 and ASME/ANS RA-Sa-2009, only minor changes were made in the internal events portion of the standard. No gaps were identified relative to the ASME/ANS RA-Sa-2009 requirements.

Based on satisfaction of the Independent Assessment process against the requirements of ASME/ANS RA-Sa-2009 and with closure of all outstanding findings associated with the internal events model, in conjunction with no gaps being identified in the comparison of the standards used for previous peer reviews, the PRA internal events model is fundamentally compliant with Regulatory Guide 1.200, Revision 2 for the scope of this application, and meets Capability Category II or above in the ASME PRA Standard (ASME/ANS RA-SA-2009). As such, the PRA internal events model can support risk-informed applications requiring Capability Category I or II capabilities, including this application.

A.2. External Events PRA Quality Statement for Permanent 15-Year ILRT Extension

External hazards were evaluated in the PNPP Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program [Reference 37]. The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose for the identification of potential plant vulnerabilities and the understanding of associated severe accident risks.

The results of the PNPP IPEEE study are documented in the PNPP Individual Plant Examination of External Events for Severe Accident Vulnerabilities Report submitted to the NRC [Reference 30]. Each of the PNPP external event evaluations were reviewed by the NRC and compared to the requirements of NUREG-1407 [Reference 38]. The NRC transmitted to FENOC on March 9, 2001 their Staff Evaluation Report of the PNPP IPEEE Submittal [Reference 39].

The PNPP PRA is a Level 1 and 2 model that includes internal events and internal floods. For external events such as fire, seismic, extreme winds and other external events, the risk assessments from the IPEEE [Reference 30] can be used for insights on changes to ILRT intervals.

A.2.1 Fire Analysis

The PNPP PRA does not include a fire model. Therefore, the results of the fire risk assessment performed for the IPEEE must be qualitatively assessed for impact of the ILRT extension on fire risk. The IPEEE fire risk analysis quantified a CDF impact by combining the frequency of fires and the probability of detection/suppression failure with the remaining safety function unavailabilities. A systematic approach was used to identify critical fire areas where fires could

fail safety functions and pose an increased risk of core damage if other safety functions are unavailable. The IPEEE CDF due to fires is $3.1*10^{-5}$/yr, with the dominant risk being fires in the control room, switchgear rooms, and specific elevations of the Control Complex, Fuel Handling, and Turbine Building [Reference 30].

A.2.2 Seismic Analysis

The PNPP IPEEE seismic risk analysis did not quantify a CDF impact, but performed a seismic margin assessment that showed that PNPP is capable of attaining shutdown conditions and maintaining these conditions for 72 hours following a review level earthquake (RLE) of 0.3g [Reference 30].

While peer reviewed, Perry's Seismic PRA model (SPRA) is not an effective model and has not been used for the ILRT extension analysis. The SPRA currently does not reflect the as-built, as-operated facility due to plant procedures and/or modifications not being in place or currently under active development. Both the underlying seismic peer review and the Independent Assessment concluded this observation [References 52 and 53]. To date, this item remains an open, unaddressed Independent Assessment finding per the previous provided report. This remaining open F&O prevents Perry from issuing that model under FENOC's process as it lacks sufficient technical adequacy in this aspect, and as such, cannot be used to support any risk application.

The seismic CDF values reported in Table D-1 of GI-199 [Reference 36] are used for estimating Seismic CDF in this calculation.

A.2.3 High Winds, Floods, and Other External Hazards

The risk of other external events such as high winds, floods, aircraft accidents, hazardous materials and turbine missiles was assessed in the PNPP IPEEE. The IPEEE assessments concluded that the impact on PNPP from these events leads to the conclusions that there are no significant events of concern [Reference 30].